STATUS OF THE CANADIAN NUCLEAR FUEL WASTE MANAGEMENT PROGRAM

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ABSTRACT

The Canadian Nuclear Fuel Waste Management Program is at the halfway point of a ten-year generic research and development phase. The major objective of this phase of the program is to assess the basic safety and environmental aspects of the concept of isolating immobilized fuel waste by deep underground disposal in plutonic rock.

The major scientific and engineering components of the program, namely immobilization studies, geoscience research, and environmental and safety assessment, are reviewed.

INTRODUCTION

Canada's radioactive waste management activities[1] include research programs on highly radioactive nuclear fuel waste[2], low- and intermediate-active waste[3-6], and uranium mine and mill tailings[7,8].

Research and development pertaining to highly radioactive nuclear fuel waste, which are the subjects of this paper, are performed within the Canadian Nuclear Fuel Waste Management Program. The program covers interim storage, transportation, immobilization, and subsequent disposal of nuclear fuel waste. The term "fuel waste" is taken to mean both used fuel discharged from CANDU™ reactors and radioactive waste that would result from recycling the used fuel, should recycling be implemented in the future.

In 1978 June, the governments of Canada and Ontario announced an agreement to cooperate in the development of technologies for the safe management and permanent disposal of Canada's nuclear fuel waste[9].

In 1981 April, the Canadian government approved a ten-year generic research and development program on nuclear fuel waste management. The objectives of this phase of the program are

(a) to assess the environmental and safety aspects of the concept of isolating immobilized fuel waste by deep underground disposal in plutonic rock;

(b) to develop the technology for storage, transportation, immobilization and disposal to the extent necessary to provide data for the assessment; to design facilities; to specify operating processes and procedures; and to demonstrate that practical technology is available for implementation of the concept;

(c) to establish the requirements, equipment, and procedures for the site characterization and selection processes for the next phase of nuclear fuel waste management; and

(d) to develop the basis for public acceptance and support through scientific and regulatory review, and public information, interaction and participation.

Under the Canada/Ontario Nuclear Fuel Waste Management agreement[9], responsibility for the development of technologies for interim storage and transportation of used fuel rests with the provincially owned utility, Ontario Hydro, while the coordination and management of the research and development program on immobilization of fuel waste and its safe disposal are the responsibility of the federal Crown Corporation, Atomic Energy of Canada Limited (AECL).

To ensure that sufficient technical expertise is available within the program, AECL has actively encouraged the participation of Canada's scientific and engineering community. Several government departments and agencies, private industry and consultants are working with AECL. In addition, faculty members of several Canadian universities have research contracts covering a wide range of topics. Over 400 scientists and engineers are contributing to the program. The administrative structure, main research and development components, participating organizations, and international cooperation are described in more detail in the Program Guide[10].

Canada has cooperative agreements with the United States of America, the Commission of the European Communities, and Sweden. These agreements provide for the exchange of data and other information on nuclear waste management, and encourage cooperation in areas of mutual interest.

At this, the halfway point in the ten-year program, it is clear that considerable progress has been made towards achievement of the program objectives. All necessary activities are in place and the remaining steps required to complete this phase of the program have been identified.

ASSESSMENT OF THE CONCEPT

The goal of the environmental and safety assessment is to assess the impact of a disposal facility on man and the environment. The assessments are being published in a series of Concept Assessment Documents. The first interim Concept Assessment Document was published in 1981[11] and the second in 1985[12,13]. The formal Concept Assessment Document, scheduled for completion in 1988, will form the basis for concept evaluation by regulatory and environmental agencies and for subsequent review at a public hearing.

The environmental and safety assessment has two major components: pre-closure assessment and post-closure assessment. Pre-closure assessment covers the period up to and including vault backfilling, sealing and closure. Post-closure assessment covers the period after the vault has been sealed and the surface facilities decommissioned.

The pre-closure assessment deals with the potential health, environmental and socioeconomic impacts of the activities: construction of a disposal facility; transportation, immobilization and emplacement of the fuel waste; backfilling and sealing of the vault; and decommissioning of the surface facilities. Ontario Hydro performs the pre-closure assessments and has completed documentation of a second interim pre-closure assessment[14-16]. Estimated occupational and public impacts were found to be within currently accepted limits for all operations. Several areas were identified where system modification could further reduce the impacts.

The post-closure assessment considers the potential long-term effects of a disposal vault and its contents on man and the environment after the vault has been sealed[17,18]. Most of the research and development activities focus on the post-closure assessment.

In 1985 February, the Canadian Radiation Protection Association met to consider criteria for nuclear waste management[19]. The meeting included representation from the Atomic Energy Control Board, Atomic Energy of Canada Ltd., and Federal and Provincial Departments of the Environment. Following the meeting, two basic criteria for acceptability of a disposal system were adopted, thus facilitating ongoing development of assessment methodology pending the promulgation of formal criteria by the regulatory and environmental agencies. The two criteria are stated as follows:

1. The estimated risk to an individual from a disposal vault during the post-closure phase should not exceed an established risk level.

2. The established probability of exceeding an established individual annual dose equivalent level should not exceed an established probability level.

The above "established" levels have yet to be decided and no conclusion has been reached as to how far in the future the criteria should apply. The first criterion is considered to be the primary criterion. The Nuclear Energy Agency of the Organization for Economic Cooperation and Development (OECD) has proposed a risk criterion for nuclear waste disposal and has suggested a level of acceptability of risk at 10^{-5} per annum[20]. Risk is a combination of the probability of receiving a radiation dose and the probability of a health effect arising from that dose.

The second criterion has been adopted to address concerns about high dose levels, even if the first criterion is met. It also enables comparison with natural background dose and current regulatory limits.

Post-closure assessment must draw on vast amounts of research information and data to estimate the overall behavior of a disposal vault. This information will concern the behavior of waste forms, containers, buffer and backfill material, the geological formation and the surface environments. In all of these aspects, there will be uncertainty and variability - uncertainty because parameters cannot be measured exactly and because their future values cannot be predicted with certainty, and variability because parameter values actually vary considerably in space and time.

SYVAC, a SYstems Variability Analysis Code for predicting radionuclide doses to man for nuclear waste disposal, was developed to perform the post-closure assessment[21]. SYVAC contains a set of submodels that represent the components of the disposal system, the vault, geosphere and biosphere. Uncertainty and variability are treated by representing the input parameters as distributions rather than as single values. The submodels and parameter distributions are derived by assimilating the results of the field and laboratory observations, which are usually interpreted by the use of detailed research models. SYVAC performs repeated deterministic calculations with parameter values sampled from their distributions in a Monte Carlo process[22].

Validation of the assessment is achieved by a combination of quality assurance on software[23], expert review, intercode comparison and comparison with field and laboratory observations, the last applying mainly to the validation of the research models. An excellent example of the validation of a research model is the comparison between prediction and observation of the water-table drawdown during construction of the Underground Research Laboratory[24].

RESEARCH AND DEVELOPMENT

A comprehensive research and development program is now firmly established[25], which provides the technology required for storage, transportation and immobilization of fuel waste, and for the construction of disposal facilities. It supports the assessments by providing recommended models and data distributions, characterizes waste forms, engineered barriers and natural barriers, and develops procedures and equipment for site characterization and selection.

Storage and Transportation of Used Fuel

Used CANDU™ fuel continues to be stored safely and economically in water-filled concrete storage bays at the nuclear generating stations. The current, installed, nuclear generating capacity in Canada is about 10 000 MW(e) and, at the end of 1985, about 400 000 used-fuel bundles, approximately 8 600 Mg, were in storage. Studies are being conducted on the feasibility and economics of two dry-storage systems - convection vaults and concrete canisters.

Experience with wet and dry storage of used CANDU fuel over the past 20 years provides confidence that interim storage is practicable for at least 50 years[26].

Ontario Hydro is developing the technology for large-scale transportation of used fuel. The reference cask design has a two-module (192-bundle) payload, rectangular geometry, and monolithic stainless steel wall construction. The heat-dissipation capabilities of the reference cask have been investigated in a full-scale simulation using electrically heated, simulated fuel bundles, and experiments have been conducted that show that little radioactivity would be released from the fuel in the event of sheath rupture during either normal or accident conditions of transport. Completion of design, construction and licensing of a full-size cask is scheduled for 1988.

Fuel Immobilization

Fuel immobilization studies involve the development of durable containment for the disposal of intact used-fuel bundles, and the characterization of used fuel as a waste form[27,28]. Studies have concentrated on simple cylindrical containers with a high-integrity, corrosion-resistant, metallic shell to isolate the fuel during its high toxicity phase. Containment systems that could offer substantially longer isolation, using materials such as ceramics, are also being studied.

Several container designs are being evaluated [29,30]. The simplest, the "stressed-shell" design, has a shell thick enough to withstand the hydrostatic pressure in a flooded vault. Others, called "supported-shell" designs, have an internal support that permits the use of thin-walled shells. The support is provided by a cast metal matrix (for example, lead), packed particulate material surrounding the fuel bundles (for example, glass beads), or a structural support (carbon steel tubes).

Prototypes of these container designs were fabricated from stainless steel or grade-2 titanium and subjected to tests in a Hydrostatic Test Facility (HTF) at pressures up to 10 MPa and temperatures up to 150°C. While short-term tests have shown all containers tested to be acceptable, a detailed structural analysis indicated that a stressed-shell container fabricated from ASTM grade-2 titanium would begin to buckle under creep deformation after about five years under a pressure of 9.4 MPa at 100°C[31]. Modelling of the structural performance of a lead-matrix design has shown good agreement with experiments[32] and voids due to casting defects have been shown not to have significant effects. Hydrostatic testing of full-scale particulate-packed and structurally supported containers has demonstrated excellent resistance to loading, with only minor deformation[33,34].

Acceptable closure welding has been achieved for grade-2 titanium using tungsten inert gas (TIG) welds[35] and development has progressed on copper container electron-beam welding[36]. Various weld-inspection techniques have been investigated, including ultrasonic techniques[37,38]. Titanium alloys are readily inspectable by ultrasonic techniques, but more development work is required for inspecting welds in copper and nickel-based alloys.

The long-term corrosion behavior of candidate container materials is being studied[39]. Grade-2 titanium has been shown to be susceptible to the initiation of crevice corrosion, but, below a critical potential, the propagation is inhibited[40]. The low levels of dissolved oxygen in deep groundwater, the use of redox buffers, and the very limited access of oxidants to the waste containers should all inhibit propagation. It has been suggested that the generation of oxidizing species at the container surface by gamma radiolysis could increase propagation; however, electrochemical experiments have shown little evidence of this so far[41]. It has been concluded that grade-12 titanium is much more resistant to localized corrosion than the grade-2 material[42], as is Hastelloy C-276, a nickel-based alloy.

Hydrogen embrittlement experiments under dynamic-strain conditions have only shown some evidence of embrittlement for grade-12 titanium at temperatures above 100°C and at highly reducing potentials[43,44]. Studies of the corrosion behavior of copper in simulated, high salinity groundwater have shown that copper is a suitable alternative to passive metals[45].

Used-Fuel Characterization

The properties of used fuel relevant to its performance as a waste form are being examined in detail, in particular the leaching and dissolution behavior. The amounts of cesium-137 and iodine-129 in the fuel-sheath gap and at the UO_2 grain boundaries have been determined as a function of fuel irradiation history. Leaching studies indicate that short-term releases in dissolution tests (termed "instant releases") result from complete removal of the gap inventory and partial release of the grain boundary fraction[46]. Used-fuel dissolution experiments at 150°C have shown that fission product and uranium releases to solution are strongly redox dependent, with lower releases occurring under reducing conditions[47]. The products of α-radiolysis of water have been shown to cause some oxidation of UO_2[48,49], although it is not yet clear if this will significantly influence the performance of used fuel as a waste form.

Waste Immobilization

Waste immobilization research is focused on developing processes and products for immobilizing the wastes arising from the recycling of used CANDU™ fuel. Process development studies are examining evaporation, calcination, vitrification, ion-exchange and off-gas treatment methods. Waste forms suitable for disposal are being developed for high-level wastes, iodine-129, carbon-14 and Zircaloy fuel bundle components.

Horizontal and vertical wiped-film evaporators have been used successfully to concentrate liquid wastes. The Waste Immobilization Process Experiment, consisting of a rotospray calciner, an electromelter and an off-gas system, and capable of producing 10 kg/h of non-radioactive sodium borosilicate glass, has completed several months of successful operation [50a,b].

Methods have been developed to remove iodine-129, carbon-14 and krypton-85 gases from process streams. Both corona-discharge and photochemical scrubbing techniques have been applied to remove iodine from air[51]. Molecular sieve adsorbents have proven satisfactory for the removal of krypton-85 from air[52]. Carbon dioxide (containing carbon-14) has been separated from air using calcium hydroxide. A number of potential hosts for iodine-129 have been examined, and it appears that the phase assemblage $Bi_2O_3 + Bi_5O_7I$ may be a satisfactory waste form for iodine-129[53]. Lead and bismuth hydroxy-carbonates may be suitable hosts for carbon-14.

Glass, glass-ceramic and ceramic waste forms are being examined in terms of their fabrication parameters and leaching performance. Temperature-viscosity relationships in the sodium borosilicate system have been studied[54], and miscibility limits have been determined for various multicomponent borosilicate glasses. The durability of aluminosilicate glasses has been examined as a function of composition, and the results indicate that these glasses have low leach rates over a wide range of composition[55].

Sphene-based glass-ceramics have been prepared by melting at 1250°C, followed by controlled crystallization, and by the use of sintering techniques at 950°C. The latter approach appears promising and may permit production of this waste form at temperatures that minimize radionuclide volatilization and eliminate refractory corrosion. Leaching studies show that sphene-based glass-ceramics are highly durable in saline groundwaters. No significant enhancement in leaching due to γ-irradiation (4 Gy/h) was observed in a 485-day test at 100°C. Leach tests of metamict sphene, produced by heavy-ion bombardment, simulating α-decay, indicate that the durability is lower than that of undamaged sphene by a factor of 5. Sphene-based glass-ceramics, however, show no such leach rate enhancement for similar irradiation conditions[56].

Vault Chemistry

The vault chemistry program is examining chemical processes that could occur in the near-field of a disposal vault. Such processes include the sorption of radionuclides by buffer materials, the hydrothermal alteration of bentonite, and the interactions of waste forms with container materials, buffer materials, groundwaters and host rock under disposal temperature and pressure conditions.

Because early assessment results suggested that iodine-129 could be one of the principal contributors to radiation dose from the disposal of used fuel, research on sorption of radionuclides in the near-field has focused on developing buffer additives to remove iodide from solution. Although several metal oxides and sulfides show an affinity for iodide, their ability to sorb iodide is greatly reduced in the presence of bentonite. This effect is due to the destabilization of metal oxy-iodides by carbonate released from the clay[57].

The diffusion of strontium, plutonium, and americium in bentonite/sand buffer materials has been studied over a range of buffer density and temperature conditions. The results show that reasonable agreement is obtained between measured diffusion profiles in clays and those calculated using a simple diffusion model[58].

Interactive tests involving waste forms (both used fuel and fuel recycle waste glasses), container materials, buffer materials, groundwaters and host rock are being performed in a series of experiments in the Immobilized Fuel Test Facility[59]. The first test, which contained 18 separate experiments, was recently terminated after 6 months of operation. These experiments are designed to aid in the development of source-term models for SYVAC, and to provide information on possible synergistic interactions among near-field components that may influence radionuclide release from waste forms.

The reactions of bentonite with various synthetic groundwaters have been studied over a broad range of temperature and pH conditions[60]. Recent studies have examined the interaction of bentonite and groundwaters with various cements that might be used in a disposal system, and results suggest that the presence of bentonite accelerates the rate of dissolution of the cements.

Disposal Vault Sealing

Disposal vault sealing studies involve the development of the buffer material (clay/sand mixture) to surround the containers, and other barriers to close the man-made openings to the surface: namely, the backfill and the plugs and grouts for shaft and borehole seals[61].

A study of the physical and chemical properties of buffer and backfill clays[62] has provided information on basic mineralogical, chemical and physical properties, and on the behavior of clays under wet-dry cycling. Bentonites are suitable as buffers because of their high swelling potentials, low hydraulic conductivities, low effective porosities and high sorption capacities for radionuclides[63-65].

Some important physical properties of candidate buffer materials have now been characterized. A compaction study[66] showed that the effective clay density in a clay-sand mixture (that is, the ratio of the mass of clay to the volume of clay and voids in the mixture) remains nearly constant for clay contents over 50 weight percent. Effective density is one of the main factors determining the effective porosity and, thus, the hydraulic conductivity and ionic diffusion properties of the material[64,67]. Swelling pressure has also been shown to be dependent on the effective clay density for one clay-sand mixture[68].

Hydraulic conductivity values measured for two candidate materials showed that sodium bentonite clay-sand mixtures have lower conductivity (10^{-11} to 10^{-13} $m.s^{-1}$) than illite clay-sand mixtures (10^{-9} to 10^{-12} $m.s^{-2}$)[69]. A model was developed to describe the factors (structure, density, water chemistry and hydraulic gradient) that determine the effective porosity of these mixtures. It was predicted that water chemistry should not significantly affect the porosity for the density values proposed for the buffer[70].

Mechanical changes can influence the effectiveness of the buffer as a thermal conductor and protective blanket around the waste container. Experimental studies have shown that shrinkage, long-term creep, drying and rewetting, and the removal of buffer material by groundwater are unlikely to prejudice the effectiveness of the buffer[71-73].

A major study on buffer and backfill engineering has provided information on procedures, schedules and costs[74,75]. The emplacement of containers was assumed to involve four stages: (i) compacting the buffer, (ii) drilling holes into the buffer, (iii) emplacing containers and capping the holes, and

(iv) backfilling. Most operations involving backfill and buffer could be carried out by conventional equipment in radiation-free conditions. However, emplacement of the container and capping of the holes would require the use of shielded or remotely controlled equipment.

Computer modelling studies have been performed to determine the effects of container and buffer geometry and the quality of the rock wall in the emplacement boreholes on diffusional transport of radionuclides from failed containers[76-80].

Immobilized Fuel Test Facility

The Immobilized Fuel Test Facility (IFTF) at the Whiteshell Nuclear Research Establishment (WNRE) provides an environment for a wide range of multicomponent experiments[81,82] in radiation fields, under temperatures and pressure conditions that simulate a vault environment. The experiments are designed to test active waste forms and materials proposed for engineered barriers. Preparation of long-term immersion experiments in passive canisters and of multi-component-systems tests are well underway. The first set of experiments was emplaced late in 1984. A typical set of experiments comprises 18 small titanium pressure vessels, each containing fuel waste, container material, buffer, groundwater and rock, loaded in one of seven concrete canisters. The experiments are run for 6 months or more at temperatures up to 200°C and at pressures up to 8 MPa.

GEOSCIENCE RESEARCH

The objectives of the geoscience program are to develop techniques suitable for site screening and evaluation, and to contribute to the general assessment of the concept of disposal in plutonic rock. The field research program is evaluating relevant physical, chemical and mechanical properties of the rock, and chemical and hydrogeological properties of the rock-groundwater system. Laboratory geochemistry programs are examining radionuclide-rock-groundwater interactions and the thermomechanical stresses that would occur in rocks near a disposal vault. These have been extensively reported in the literature[83].

Field Research

Deep exploratory drilling and surface mapping are being carried out to obtain the data necessary to develop site-selection criteria. Various new and existing techniques are being evaluated at five field research areas. The Chalk River, Atikokan and Whiteshell research areas are in granitic rock, whereas those at East Bull Lake and Overflow Bay are in gabbroic rock.

Methods of correlating subsurface fracture characteristics to surface fractures are required in order to select a suitable disposal site. Airborne and surface geophysical monitoring techniques and air photo lineament analysis are being used to predict subsurface fracture conditions, and drilling is being carried out to test the predictions[84].

The mineralogy, geochemistry and relative chronology of fracture fillings in plutons are being studied. Uranium disequilibrium and oxygen-18 analyses of calcites, clays and goethites found in open fractures in granite indicate that they have reacted with waters during the last 250 000 years, suggesting that these minerals are the products of recent rock-water interactions[85].

Brackish and saline waters have been found at depths of greater than 300 m at all research areas, with salinities ranging from 3 g/L total dissolved solids in gabbro to 40 g/L in granite. If correction is made for dilution due to surface waters, concentrations of over 140 g/L are estimated for a depth of 1 km in granite[86].

Underground Research Laboratory

A hydrogeological monitoring network has been established at the Underground Research Laboratory (URL), located north of the WNRE site, to determine the effects of the URL construction on the groundwater system. Groundwater inflow rates, water chemistry variations, and piezometric levels were monitored continuously as the URL shaft excavation proceeded. The shaft has been completed to the target depth of 240 m. Models were developed, using data from the drilling program done before shaft sinking, to predict the response of the groundwater system to the excavation of the URL. Comparison with the data obtained during excavation shows good agreement on water inflow rates and excellent agreement on the extent and degree of water-level drawdown[87].

An extensive program of experiments is planned for the URL. This includes determination of the rock response to thermal and mechanical loading, testing of buffer and backfill emplacement techniques, and borehole, drift and shaft sealing experiments.

The URL is the first subsurface facility to be excavated below the water table in undisturbed plutonic rock. Although the URL was designed to meet the needs of the Canadian Nuclear Fuel Waste Management Program, the facilities will be made available to other nations, groups or organizations. This involvement may be on the basis of shared costs for experiments of mutual interest, or on a commercial basis.

An agreement is being negotiated between AECL and the United States Department of Energy that would include cooperative experimental activities in the URL. If negotiations are successful, the shaft would be extended to a depth of 455 m.

Vault Engineering Studies

A variety of alternative designs for a disposal vault have been examined, involving single-level, multi-level and long-hole emplacement concepts[3]. In addition, a comprehensive engineering study examined acquisition, transportation, emplacement methods, and costs of buffer and backfill materials for a disposal vault[74-76].

For a used-fuel disposal vault, the results show that construction costs would be significantly higher for a multi-level vault than for a single-level vault, due to the need for lower loading density in the former to meet thermal constraints. However, a multi-level vault for immobilized fuel recycle waste is feasible, since the cost per container would be only 11% higher than for a single-level vault. The option of constructing a multi-level vault may be valuable, if rock quality or pluton size limits the area available for a vault. Potential problems with short-term container retrieval and buffer emplacement quality control would appear to rule out a long-hole disposal method.

A comprehensive conceptual engineering study is currently being commissioned for the overall concept of immobilization and disposal of nuclear fuel wastes [88].

Geochemistry

The processes that influence migration of radionuclides in fractured plutonic rock are being studied to provide the basis for detailed radionuclide transport models. Short-term laboratory and field experiments provide information on the parameters affecting radionuclide sorption and transport. Field geochemistry studies have examined the end results of natural geological processes, to obtain data on rock alteration and radionuclide migration.

Significant progress has been made in the development of suitable chemical equilibrium codes and compilation of associated thermodynamic data for calculating groundwater composition in equilibrium with mineral assemblages. Work on an internally consistent, thermodynamic data base for minerals in the system Na_2O-K_2O-CaO-MgO-FeO-Fe_2O_3-Al_2O_3-SiO_2-TiO_2-H_2O-CO_2 is almost complete[89]. A thermodynamic data base for neptunium has been completed[90], and heat capacity data for a number of actinides, fission products and groundwater species have been compiled [91].

The Tc(IV)/Tc(III) redox couple has been studied in aqueous bicarbonate solution[92]. It appears that Tc(IV) carbonate complexes will be the most important species in reducing groundwaters. Sorption studies, however, indicate that these complexes are strongly sorbed by iron oxides[93].

Chemical stripping techniques have been used to determine the minerals responsible for radionuclide sorption. Iron oxide and oxyhydroxide surface coatings on minerals appear to play a major role in radionuclide sorption[93].

A technique has been developed to measure pore space in rocks, and this has been applied to highly altered rock adjacent to a water-bearing fracture. The porosity of the altered rock is a factor of two greater than that of unaltered rock[94].

Matrix diffusion of non-reactive species in granite has been studied, and this retardation mechanism is being evaluated for possible inclusion in the geosphere submodel used in SYVAC.

Natural analog studies provide information that can be used in predicting the behavior of radionuclides in rock surrounding the disposal vault. Uranium deposits in Northern Saskatchewan are being examined as analogs of a nuclear fuel waste disposal vault. Although water-rock interactions have occurred in the ore bodies a number of times since the deposits formed almost a billion years ago, reducing conditions limited uranium migration in the clay surrounding the deposits to less than 5 metres[95].

ENVIRONMENTAL RESEARCH

The environmental research program is studying pathways by which radionuclides may migrate in the biosphere, and the ultimate radiation dose to man due to these various exposure routes. Potential exposure routes being examined include ingestion of well water, terrestrial plants and animals, surface water, and aquatic plants and animals, inhalation of air, and external exposure from soil, air and water.

The location of groundwater discharge zones relative to lakes and rivers, and the discharge rates, will determine the environmental concentrations of the radionuclides and the predominant exposure pathway. Airborne thermal imagery appears to be a promising method for locating such discharge zones [96]. Identification of submerged discharge zones is being examined using a lakebed drag with temperature and conductance probes.

Studies of dispersion in contaminant plumes are providing data on the importance of this phenomenon in controlling radionuclide concentrations in sediments. Results show that aquifer stratification is the dominant control on large-scale dispersion, and that transverse dispersion within strata is a factor of 5 to 10 lower than longitudinal spreading[97]. Ground-probing radar has proven valuable in defining stratigraphic features that may influence dispersion in unconsolidated sediments[96].

The relative contributions of groundwater flow and surface runoff to streamflow could influence radionuclide concentrations in surface waters. Studies using deuterium as a tracer indicate that, even in the spring, streamflow is still predominantly composed of groundwater[96]. Sedimentation-resuspension processes in surface waters are also being examined, as these influence radionuclide concentrations in lakes and rivers.

Organic soils occur in low-lying areas in the Canadian Shield where groundwater discharge is expected to occur, and they may have a dominant effect on radionuclide migration in the biosphere. Migration of a variety of radionuclides in moss and peat cores is being studied in field tests. Sorption of a number of radionuclides by soils is being examined for both aerobic and anaerobic conditions. The long-term dynamics of chemical elements in Shield soils are being evaluated through studies of weathering and podzolization[96].

The uptake of a number of radionuclides by plants has been studied over a soil concentration range of three orders of magnitude. Results show good confirmation of a linear concentration ratio model[96].

Potential atmospheric exposure pathways are being studied, including soil suspension by wind, and gaseous and particulate suspension due to forest fires and burning wood for fuel.

A food chain model, LIMCAL, has been incorporated into SYVAC to calculate dose equivalents for man. LIMCAL includes all the major ingestion pathways leading to man, including terrestrial, fresh-water and salt-water food types, as well as man's and animal's drinking water[98]. The importance of inhalation pathways for both animals and man has been evaluated, and inhalation by animals was shown to be a potentially important pathway, particularly for some actinides[99]. A review of the importance of the soil ingestion pathway for animals indicated that, for cattle, this exposure pathway is much more important than root uptake by plants for many radionuclides[100].

REVIEW PROCESS AND SCHEDULE

In 1981 August, the governments of Canada and Ontario issued a statement describing the evaluation process, the roles and responsibilities of the environmental and regulatory agencies, and the involvement of the public[101]. The evaluation process, which will start in 1988, will involve a regulatory and environmental review, a full public hearing and, in 1991, a decision by the two governments on the acceptability of the concept. In the regulatory and environmental review, the Atomic Energy Control Board will act as the lead agency, assisted by the federal Department of the Environment and the Ontario Ministry of the Environment. The public hearing will be held under the auspices of the Canadian government.

Major efforts have been made to involve the public throughout the program, with speaking engagements, discussion group meetings, and invitations to review assessment documentation. A comprehensive public consultation program is now being launched.

An independent Technical Advisory Committee, established in 1979, provides an ongoing scientific review of the program. The membership of the Committee is drawn from candidates nominated by professional societies throughout Canada, thus ensuring its independent status. The Committee advises AECL on the extent and quality of the program and interprets and evaluates it for the scientific and technical community and the general public. The Committee also makes constructive criticisms, suggestions and recommendations concerning the various components of the program. The Technical Advisory Committee has issued six annual reports available to the public (see for example, reference 102).

In the second half of the ten-year program it now remains to consolidate the site screening and evaluation methodology, the development of the technology for immobilization and disposal, and the scientific basis for the concept assessment, so that the regulatory and environmental authorities, the public and the various levels of government can reach decisions on the acceptability of the concept.

CONCLUSIONS

The basic components of the Nuclear Fuel Waste Management Program, including research and development related to used-fuel storage and transportation, fuel waste immobilization, site characterization and performance assessment modelling have been described. Preliminary performance assessment results support the view that disposal in plutonic rock appears to be a promising approach for long-term management of nuclear fuel wastes. The program remains on schedule, and research is directed at the completion of documentation on concept assessment for submission to the Governments of Canada and Ontario in 1988.

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