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THE CANADIAN NUCLEAR FUEL WASTE MANAGEMENT PROGRAM

LE PROGRAMME CANADIEN DE GESTION DES DECHETS DE COMBUSTIBLE NUCLEAIRE

T. E. Rummery, E. L. J. Rosinger

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Etablissement de recherches nucléaires de Whiteshell

Pinawa, Manitoba R0E 1L0

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by

T.E. Rummery and E.L.J. Rosinger

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LE PROGRAMME CANADIEN DE GESTION DES DÉCHETS DE COMBUSTIBLE NUCLÉAIRE

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RÉSUMÉ

Le Programme canadien de gestion des déchets de combustible nucléaire en est à la quatrième année d'une phase de recherche et développement génériques de dix ans. L'objectif de cette phase du programme est d'évaluer la sûreté de base et les aspects écologiques du concept d'isolation des déchets de combustible immobilisés dans une enceinte d'évacuation aménagée à grande profondeur dans la roche plutonique.

Le rapport décrit les éléments scientifiques et techniques principaux du programme, c'est-à-dire, les études de l'immobilisation, la recherche géoscientifique et l'évaluation de l'environnement et de la sûreté. On traite brièvement du financement du programme, du calendrier des travaux et des processus externes d'examen associés à ce programme.

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ABSTRACT

The Canadian Nuclear Fuel Waste Management Program is in the fourth year of a ten-year generic research and development phase. The 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 described. Program funding, scheduling and associated external review processes are briefly outlined.

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CONTENTS

1. INTRODUCTION

Canada's radioactive waste management activities [1] include research programs pertaining to highly radioactive nuclear fuel waste [2], low- and intermediate-level 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 involves research and development of technologies for the 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 of 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]. Since that time, the program has grown to become truly national, with participation from federal and provincial government departments, from industry, and from the academic community.

In 1981 April, the Canadian government approved a ten-year generic research and development program on nuclear fuel waste management. During this ten-year concept assessment phase, 1981-1990, the program is expected to provide the data necessary to assess the concept of deep underground disposal. The process by which the concept will be evaluated through regulatory review and a public hearing was defined in a Canada-Ontario joint statement, issued in 1981 August [10].

The objective of the Canadian Nuclear Fuel Waste Management Program is to ensure that there will be no significant adverse effect on man or the environment from nuclear fuel waste at any time. The objective incorporates two basic principles: safety - the management of radioactive waste so that the hazards are negligible, and responsibility - the management of radioactive waste so as to eliminate, or minimize, the involvement and concern of present and future generations. To achieve the program objective, it has been proposed that the suitably immobilized fuel waste be emplaced deep underground in stable plutonic rock within the Canadian Shield.

The program for the management of nuclear fuel waste has four major components:

- interim storage of used fuel,
- transportation of used fuel,
- immobilization of used fuel and fuel recycle waste before disposal, and
- permanent disposal of the immobilized waste.

+ Also called irradiated or spent fuel.

* Canada's natural-uranium-fuelled, heavy-water-moderated and -cooled reactor (CANada Deuterium Uranium).

Responsibilities for the program components are defined in the Canada/Ontario Nuclear Fuel Waste Management agreement [9]. Under this agreement, 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).

CANDU reactors operate on a natural uranium, once-through fuel cycle. However, technologies are being developed for the immobilization of both used fuel and fuel recycle waste so that either can be disposed of in the future, as requirements demand. Current used-fuel storage methods are adequate for decades, and additional interim storage can readily be provided at the reactor sites [11]. Thus, commercial disposal is not a near-term requirement.

2. FUNDING AND PARTICIPANTS

Funding for the immobilization and disposal aspects of the program is provided by the Canadian government. For the ten-year period 1981-1990, the Canadian government has approved an average annual funding of $29 million (1981$) for generic research and development. Ontario Hydro allocated $1 million in 1983 for technical assistance in the immobilization and disposal components of the program, in addition to directing and funding the storage and transportation components.

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 [12].

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 valuable criticisms, suggestions and recommendations concerning the various components of the program. The Technical Advisory Committee has issued four annual reports [13-16], which are available to the public.

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.

3. 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 7500 MWe. At the end of 1983, about 300 000 used-fuel bundles (weighing approximately 7000 Mg) were in storage, after producing about 290 billion kWh of electricity.

After storage for a few years in water-filled bays, used-fuel heat production decreases sufficiently to permit dry storage with passive cooling. 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 [17].

The conceptual design of a dry-storage convection vault includes an array of vertical, concrete, metal-lined tubes containing the used-fuel bundles in an above-ground concrete structure [18]. Cooling is provided by natural air convection. A preliminary thermal analysis of various configurations indicates that the maximum fuel cladding temperature will not exceed 165°C, well within the temperature limit of 200°C considered safe for the long-term integrity of fuel cladding in a dry environment. Experiments are underway to assess the long-term durability of used fuel during storage in concrete canisters. Preliminary results indicate that storage in concrete canisters for up to five years in dry or moist air does not lead to deterioration of used fuel or radioactivity release.

Ontario Hydro is developing the technology for large-scale transportation of used fuel. The reference cask design includes a two-module (192-bundle) payload, rectangular geometry, and monolithic stainless steel wall construction. Completion of design, construction and licensing of a full-size cask is scheduled for 1988.

4. FUEL WASTE IMMOBILIZATION

Immobilization research and development are concerned with the man-made components of the nuclear fuel waste-disposal system. The research includes the characterization of, and development of, durable containers for both used fuel and immobilized fuel-recycle waste. Another important aspect entails development of the backfill, buffer and sealing materials for a nuclear fuel waste-disposal vault.

4.1 FUEL ISOLATION

Fuel isolation 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 [19,20]. The used fuel is itself resistant to dissolution under chemically reducing conditions and appears to be a viable waste form for disposal. Studies are 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 [21,22]. 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 at pressures up to 10 MPa and temperatures up to 150°C. Preliminary results indicate that stresses generated in the containers are acceptable and do not impair container performance.

The long-term corrosion behaviour of candidate container materials is being studied [23]. For candidate passive metals, i.e., titanium and nickel-based alloys, crevice corrosion is the process most likely to limit container lifetime. For copper, the uniform corrosion rate is expected to depend on the supply of oxidants. Studies of the corrosion behaviour of copper in simulated high-salinity groundwater have shown that copper is a suitable alternative to passive metals [24].

4.2 USED-FUEL CHARACTERIZATION

Determination of the leaching and dissolution properties of used UO_2 fuel constitutes the major part of the fuel characterization program. During the past two years, emphasis has been on estimating the fractions of cesium-135 and iodine-129 that are rapidly released from the gap between the fuel and the sheath during the early stages of used-fuel dissolution. The gap inventory of cesium-135 and iodine-129 at the time of fuel discharge from the reactors is now estimated to be about 2.2%. Recent studies have shown a correlation between fuel power history and fuel leaching properties [25].

Preliminary experiments to investigate the effects of gamma and alpha radiolysis of groundwater on UO_2 dissolution are underway. After 485 days in a gamma-radiation field at 100°C, UO_2 samples in groundwater became slightly more oxidized than did non-irradiated samples. The presence of clays reduced the oxidation effect. Experiments with an alpha source with a strength between 0.037 and 3.7 MBq (1 and 100 μCi) provided no electrochemical evidence for UO_2 oxidation. However, with a 37-MBq (1-mCi) alpha source, electrochemical results suggested that surface oxidation of UO_2 can occur. This is probably due to the reaction of radiolytically produced radicals, or hydrogen peroxide, with the UO_2 surface.

4.3 WASTE IMMOBILIZATION

The objective of the waste immobilization studies is to develop processes and products for immobilizing the waste that would arise if the fuel from CANDU reactors were recycled [20,26]. Glasses, ceramics and glass-ceramics are being evaluated as possible waste forms.

An in-can melting technique, involving a second container into which molten glass can be poured, is being evaluated. Since in-can melters have poor heat transfer and slow rates of evaporation, a wiped-film evaporator is being tested to concentrate the waste prior to vitrification. A Waste Immobilization Process Experiment facility, consisting of a rotospray calciner and a ceramic electromelter, designed to produce 10 $kg.h^{-1}$ of sodium borosilicate glass, has been constructed, and commissioning is now underway.

The behaviour of glass waste forms and their durability in the hydrothermal environment anticipated in a disposal vault are being studied. A recent survey of borosilicate waste glasses [27] showed that durability increases with increasing SiO_2, Fe_2O_3 or Al_2O_3 content, but decreases with Na_2O or K_2O content. Sodium aluminosilicate glasses [28] have a low, relatively constant leach rate (less than 10^{-9} $kg.m^{-2}.s^{-1}$) within a wide composition range. The leaching mechanism involves preferential release of alkali or alkaline earth ions by ion diffusion, creating an aluminosilicate-rich surface layer, and the concurrent, but slower, dissolution of the aluminosilicate layer. Eventually, "steady-state" dissolution occurs, when the rate of alkali ion diffusion through the leached surface layer equals the rate of silicate dissolution at the surface.

The ceramic waste forms being considered contain sphene ($CaTiSiO_5$). Calculations indicate that sphene should be stable in groundwaters typical of the Canadian Shield (high Ca^{2+}, low SO_4^{2-} and CO_3^{2-}) in the temperature range 25 to 150°C. Three types of sphene-based matrices are being studied: natural minerals, ceramic pellets formed by pressing and sintering, and glass-ceramics formed by melting and controlled crystallization of the system Na_2O-Al_2O_3-CaO-TiO_2-SiO_2. Experimental studies of the hydrothermal behaviour and surface modification of natural and synthetic sphenes in groundwater show little or no selective leaching of the matrix elements or simulated radionuclides, provided sphene is within its stability field [29,30]. In deionized water, Ca^{2+} and Si^{4+} are selectively leached, leaving a thin surface layer composed of highly insoluble TiO_2 (anatase or brookite). For the glass-ceramic, consisting of sphene crystallites in an aluminosilicate glass matrix, the release of radionuclides depends on the relative dissolution rates of the glass and ceramic phases, and also on the partitioning of each radionuclide between the two phases [31]. Solution and surface analyses indicate that in most cases the glass phase leaches preferentially. For the glass-ceramic, irradiation does not lead to an increase in leach rate in distilled water or brine at 100°C [32].

4.4 DISPOSAL VAULT SEALING

Disposal-vault sealing studies involve the development of the buffer material surrounding the containers, and other barriers that close the man-made openings to the surface, namely, the backfill and the plugs and grouts for shaft and borehole seals [33].

A study of the physical and chemical properties of buffer and backfill clays [34] has provided information on basic mineralogical, chemical and physical properties, and on the behaviour of clays under wet-dry cycling [35]. A comprehensive review of phyllosilicate clays has indicated that bentonites are suitable as buffers because of their high swelling potentials, low hydraulic conductivities, low effective porosities and high sorption capacities for radionuclides [35-38].

A study of the thermal properties of clay and crushed granite mixtures has shown that thermal conductivity increases rapidly when the clay content falls below 30 percent [39]. Experimental work on the hydrothermal stability of bentonite materials (at $pH \leq 6$) has indicated that the conversion of smectite (the major component of bentonite) to illite proceeds very slowly below 150°C, even in the presence of K^+. Recent reviews of smectite stability [38-40] suggest that smectite will be stable under disposal-vault conditions.

A major study on buffer and backfill engineering has provided information on procedures, schedules and costs [41-45]. The emplacement of containers was assumed to involve four stages: (i) compacting the buffer (either in-hole or in-room), (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. Capital, operating and maintenance costs were estimated for the two emplacement configurations (in-room and in-hole), two waste forms and three container types.

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 [46-50].

4.5 THE IMMOBILIZED FUEL TEST FACILITY

The newly constructed Immobilized Fuel Test Facility (IFTF) at the Whiteshell Nuclear Research Establishment provides an environment for a wide range of multicomponent experiments [51,52] designed to characterize radioactive waste forms and materials proposed for engineered barriers under conditions that could exist in a disposal vault. Seven concrete canisters for experiments with highly radioactive materials have been constructed and instrumented. The first set of experiments was recently emplaced and the complex data acquisition systems were connected. Experiments in the IFTF will last for up to two years.

5. GEOSCIENCE RESEARCH

The emphasis of the geoscience research is on the evaluation of large plutonic rock masses in the Canadian Shield as potential hosts for immobilized nuclear fuel waste [53-55]. The aim of the research is not to assess specific sites but to contribute to a generic assessment of the concept of disposal deep in stable plutonic rock. The research involves quantifying the in situ mechanical, physical and chemical properties of plutonic rock, the physical and chemical characteristics of groundwater, the nature of groundwater/rock/radionuclide interactions, and the thermal and mechanical stresses that would be created in the surrounding rock by the waste.

5.1 FIELD RESEARCH

Deep exploratory drilling and detailed surface mapping are being carried out at designated field research areas in the Canadian Shield. The areas at Chalk River and Atikokan, Ontario, and Whiteshell, Manitoba, contain granitic rocks, while those at East Bull Lake and Overflow Bay, Ontario, contain gabbros.

The research area near Atikokan has been chosen as the site of a regional Flow System Study to be carried out over the next eight years. The purpose of the study is to improve our understanding of groundwater flow in the Canadian Shield and to identify the factors that control the pattern and extent of the flow systems, and associated groundwater chemistry. The study area is about 20 km x 20 km and includes the Eye-Dashwa Lakes granitic pluton and a large part of the surrounding rock. Geological mapping of the Eye-Dashwa Lakes pluton and surrounding rock was carried out to place the pluton in a regional context. Surficial deposits were also mapped, and the interaction between the shallow groundwater table and the deeper flow systems was investigated. Surface water chemistry surveys located electrical conductivity anomalies (up to 300 $\mu S \cdot cm^{-1}$) that could indicate local zones of recharge or discharge. Sonar surveys of several lakes provided information on the water depth and on the sediments between the bedrock and the lake bottom.

At the East Bull Lake research area, detailed geophysical surveys were made of the gabbro-anorthosite pluton [56]. The results indicated that both airborne and ground geophysical methods have good predictive capabilities. Estimates of the pluton thickness based on gravity [56] and aeromagnetic data [57] were in the range 400 to 800 m, and magnetotelluric surveys [58] suggested the presence of a highly conducting layer at a depth of about 800 m. Subsequent drilling showed that the gabbro-anorthosite layer is about 770 m thick. There was also a good correlation between major faults mapped in the field [59-61] and conductors determined by very low frequency - electromagnetic survey (VLF-EM).

Regional groundwater sampling indicated that saline groundwaters occur at depth throughout the Canadian Shield in fractures in plutons not associated with metallic ore mineralization. The origin of these waters is not yet fully understood, and has been variously attributed to fossil sea water, intense rock-water interaction, and leakage from the Paleozoic cover [62].

5.2 UNDERGROUND RESEARCH LABORATORY

The Whiteshell research area is situated on the Lac du Bonnet batholith, a large granitic body in southeastern Manitoba. This research area is the site of the Underground Research Laboratory (URL), which is being constructed below the water table in a previously undisturbed portion of the batholith. The URL project has been underway since 1979 [63,64], when field studies commenced to identify a suitable study area and location for the laboratory. In 1980, surface and mineral leases for 21 years were obtained on 3.8 km^2 of Manitoba Government crown land 15 km northeast of the Whiteshell Nuclear Research Establishment.

The objectives of the URL project are to study the correlation between surface and subsurface features, hydrogeological and geochemical systems in plutonic rock, excavation damage in rock, the effect of heat on plutonic rock (including the effect on mass transport), and the effect of heat on the buffer/backfill/rock interactions.

Comprehensive geological, geophysical and hydrogeological investigations of the URL lease area are being carried out. Numerous geophysical surveys were performed in boreholes at the URL area [65,66] and three major subhorizontal fracture zones have been identified. Recently, activities have focussed on defining in detail the hydrogeological conditions, particularly in the immediate vicinity of the shaft and underground excavation. A network of instrumented boreholes has been established to provide baseline data on pre-construction hydrogeological conditions and to measure changes caused by the excavation. Groundwater levels will be recorded continuously in about 75 groundwater monitoring locations. Predictions of changes in groundwater systems by several independent hydrogeological modelling groups will be compared with the groundwater system perturbations measured during and after excavation.

Following the selection of the URL shaft location in 1982, the surface and underground facilities were designed (Figure 1). Surface facilities are now completed and shaft excavation began in 1984 May. The URL will be ready for operation in 1986. The underground facilities will have a 255-m deep, rectangular access shaft, a ventilation raise, and a test level with several experimental rooms.

5.3 GEOCHEMISTRY AND APPLIED CHEMISTRY

The objective of the geochemistry and applied chemistry research is to quantify the chemical and physical interactions that occur between radionuclides and the geological materials lining water-bearing fractures in plutonic rock. These interactions can prevent or retard migration of radionuclides from the deep underground vault to the biosphere. Understanding these processes is essential for assessing the effectiveness of the geosphere as a barrier to radionuclide migration.

Chemical reactions between groundwater and rock lead to changes in groundwater composition, and the physical and chemical characteristics of the materials lining the rock fractures are also altered. The chemical and physical properties of rock and groundwater affect the sorption, diffusion, and migration of radionuclides [20]. Results of routine static sorption studies of radionuclides on rock samples from research areas confirmed an earlier observation that sorption increases with increased mafic mineral content of the rock [67]. However, dynamic sorption studies on granite and gabbro did not show this effect, and further studies are planned.

Geochemical processes are often very slow, and kinetic effects have to be taken into account when extrapolating results from laboratory and

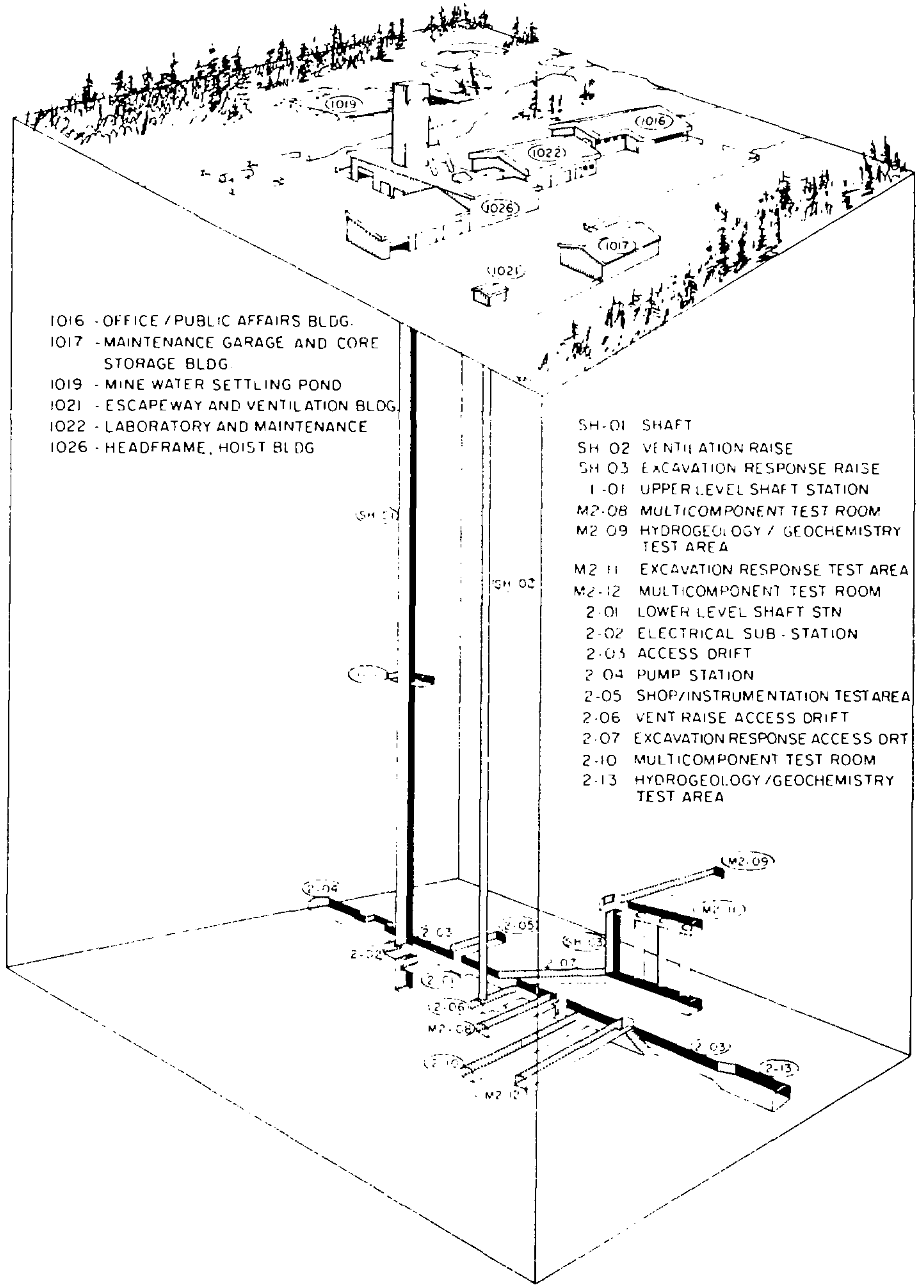

FIGURE 1: Underground Research Laboratory Design Layout

field experiments to long time periods. Studies of natural geological processes that have been in progress for long periods are providing valuable information for predicting radionuclide behaviour. Examinations of the geological records that exist in and along groundwater-bearing fractures in plutonic rock, and of geological analogues to a disposal vault, such as naturally occurring uranium deposits, are being used to assess the behaviour of radionuclides in the geosphere. In two well-defined uranium deposits in northern Saskatchewan, the uranium was observed to have migrated less than 5 m into the clay surrounding the ore body over the last billion years.

6. ENVIRONMENTAL AND SAFETY ASSESSMENT

The goal of environmental and safety assessment is to use the results of research and development 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 [68-70], and the second will be issued in late 1984. The formal Concept Assessment Document, scheduled for completion in 1988, will form the basis for concept evaluation by regulatory and environmental agencies and 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, the surface facilities decommissioned and the surface environment restored.

The pre-closure assessment deals with the potential health, environmental and socioeconomic impacts of 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. It involves analysis of radiological pathways, impacts on the biophysical environment, the use of natural resources, socioeconomic impacts, occupational safety, public safety, security, and safeguards.

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 and the surface environment restored [71,72]. The assessment is based on a systems approach, and draws together information from all parts of the program.

The systems assessment is performed using a computer program, SYVAC. SYVAC, for Systems Variability Analysis Code, is a stochastic code developed to address parameter variability and uncertainty in the long-term performance assessment of a disposal system [73,74]. It contains an executive driver routine and a set of submodels representing the components of the disposal system, i.e., the vault, the geosphere, and the biosphere. The first version of the code, SYVAC1, has been applied to preliminary assessments of used-fuel disposal in plutonic rock [70] and under the seabed [75]. The second version, SYVAC2, contains over 1100 parameters and deals with the potential consequences of more than 60 different contaminants.

The vault submodel contains descriptions of three major processes: container failure, release of radionuclides from the waste form, and radionuclide migration through the buffer. The probablity of container failure has been derived from experimental data on the rate of uniform corrosion of a thin-walled titanium container. Release of radionuclides from the used UO_2 fuel is described by a two-step release model, containing a parameter for an "instant release fraction" for radionuclides present in gaps between fuel and sheath, and parameters describing the congruent dissolution of the UO_2 matrix [76]. The release of radionuclides from the glass waste form is described by a similar model, with a short-duration "leach fraction" parameter and a long-term congruent dissolution parameter known as the "effective glass solubility" [77]. Migration of radionuclides through the buffer in the vault is described by a time-dependent, one-dimensional equation containing terms for convective transport, diffusive transport and sorption.

In the geosphere submodel, the migration of radionuclides through the geosphere is represented by a one-dimensional transport equation, containing terms describing convection, dispersion, sorption and radioactive decay.

The biosphere submodel calculates concentrations of radionuclides in air, soil, lake and well compartments. The calculated radiation dose includes that received from ingestion, inhalation and external exposure. For dose due to food ingestion, the biosphere submodel uses a version of LIMCAL, a food-chain model based on energy requirements [78], and for the dose due to the ingestion of drinking water, the submodel assumes the source of drinking water to be a fresh-water lake or a well. For the dose due to inhalation, the submodel considers the inhalation of radon gas and of particulates containing radionuclides. For the dose due to external exposure, the submodel considers radionuclide concentrations in the air, soil, lake and well compartments.

For the preliminary assessment of used-fuel disposal with SYVAC2, 2086 simulations were performed [20]. The results, shown on Figure 2, are plotted as frequency of consequence versus consequence. Consequence was defined as the maximum annual dose equivalent to a member of the most exposed group in the first million years after disposal. In the majority (67%) of the simulations, the consequence was zero (a dose less than 10^{-10} $mSv.a^{-1}$). Of the remaining 33%, about 32% showed consequences between zero and background, and 1% showed consequences greater than that from natural background radiation.

Detailed examination of the results indicated that ingestion is the predominant pathway resulting in dose to man and that contributions by the inhalation and external pathways are small by comparison. A pathway contributing to high consequences was ingestion of well water; all scenarios with consequences exceeding 0.1 $mSv.a^{-1}$ involved the well-water pathway, but not all scenarios with the well-water pathway led to high consequences. The results further indicated that the geosphere is an excellent barrier and that iodine-129 and technetium-99 are the only radionuclides contributing significantly to consequences.

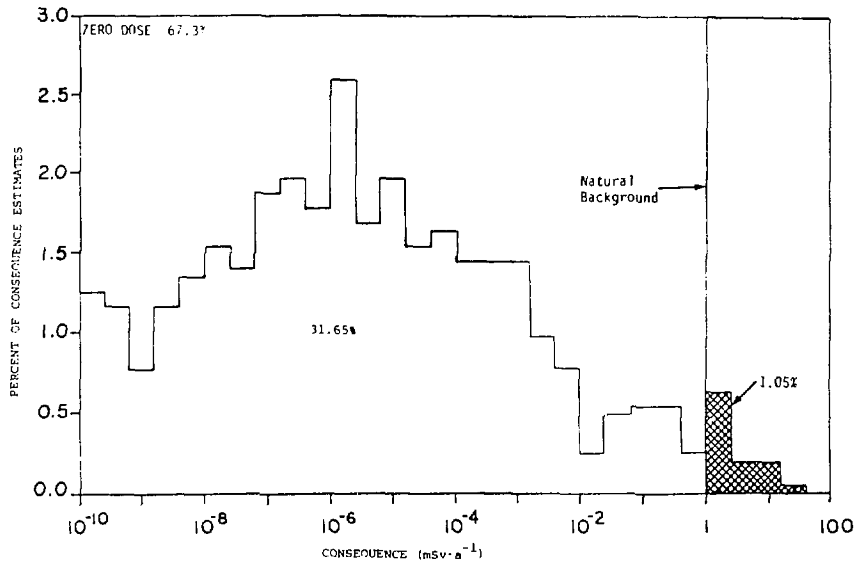

FIGURE 2: Histogram of Percent of Consequence Estimates Versus Consequence for 2086 Simulations Using SYVAC2

7. REVIEW PROCESS AND SCHEDULE

The Canadian Nuclear Fuel Waste Management Program is in the fourth year of a ten-year generic research and development phase. The objective of this phase is to assess the basic safety and environmental aspects of the concept of isolating immobilized fuel waste by deep underground disposal in plutonic rock. Laboratory and field studies provide data that will be used to assess the effectiveness of various natural and engineered barriers in preventing, or minimizing, the movement of radionuclides. Development of the methodology to make such an assessment is a vital part of the program.

During the later stages of the generic research and development phase, a formal environmental and safety assessment of the disposal concept will be presented to regulatory and environmental agencies for review and evaluation. 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 [10]. The evaluation process, which will start about 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.

The Canadian government has stated that site selection for a disposal vault will not be undertaken until the concept has been evaluated and accepted [10]. It is expected that a demonstration vault would operate for some years before a full-scale commercial vault for disposal of nuclear fuel waste is built.

Public knowledge, understanding and acceptance are important to the success of the program, and considerable emphasis is being placed on providing full and open public information on the progress of the program.

8. CONCLUSION

The Canadian Nuclear Fuel Waste Management Program is now firmly established, with a funding level consistent with its technical requirements and a schedule that permits a careful and rigorous development of the concept of disposal of immobilized fuel waste deep underground in stable plutonic rock within the Canadian Shield. Research incorporating all major components of the program has been underway for four years, and the program is proceeding on schedule and within budget. The review and evaluation process, which will start about 1988, has been defined. It is expected that the governments of Canada and Ontario will be able to make a decision on the acceptability of the concept by 1991.

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