

Waste



UNITED STATES
NUCLEAR REGULATORY COMMISSION

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MEMORANDUM TO: Michael J. Bell, Chief
ENGB/DWM/NMSS

THRU: Richard A. Weller, Section Leader *R. A. W.*
Engineering and Material Section
ENGB/DWM/NMSS

FROM: Tae M. Ahn, Materials Engineer *T. M. Ahn*
Engineering and Material Section
ENGB/DWM/NMSS

SUBJECT: TRIP REPORT ON THE 7TH ANNUAL INTERNATIONAL HIGH-LEVEL
RADIOACTIVE WASTE MANAGEMENT CONFERENCE AND EXPOSITION,
APRIL 29 TO MAY 3, 1996, LAS VEGAS, NEVADA

PURPOSE:

The purpose of this trip would: 1) update my knowledge in high-level waste (HLW), 2) be useful in my current participation in the HLW disposal program, and 3) enable me to exchange this knowledge with other participants to establish more objective views.

SUMMARY:

The conference began with the opening plenary session on high-level waste management. The speakers in this panel discussion were D. Dreyfus of the U.S. Department of Energy (DOE); H. Thompson of the Nuclear Regulatory Commission; S. Platt of the Nuclear Energy Institute (NEI); and W. Wells and D. Ferraro, Sr. of the University of Nevada, Las Vegas. Viability assessments of the Yucca Mountain project will be made by 1998, and the suitability of Yucca Mountain site will be determined by 2002. The NRC licensing is considered a critical path. Current NRC efforts include resolution of Key Technical Issues.

The second plenary session was on the role of international and national regulations assuring safe transport of radioactive materials. The participants of this session were W. Lake and his associates of DOE, D. Blackman of the United Kingdom-Department of Transportation, L. Hendricks of NEI, and R. Fisk of Edlow International. The transportation of enriched fuel from research reactors was addressed as an important safeguards issue for nuclear nonproliferation. Research reactors are in operation throughout the world.

The third plenary session discussed prospects and perspectives on nuclear waste management. D. Dreyfus of DOE, N. Johnson, congressional legislative assistant, J. Phalen of NEI, E. George of Iowa Utilities Board, and W. Arthur of South Carolina Public Utility Commission participated in the discussion. On-site storage and subsequent disposal was discussed. Diversified opinions

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about the on-site storage were presented considering economic and social constraints.

I attended sessions on criticality issues, microbial activity research, advanced technologies and engineering concepts, total system performance assessment, waste package behavior, waste form behavior, dose and biosphere modeling, and near-field performance assessment. Because many issues were covered in several sessions, the following summary is made for issues rather than sessions.

In the criticality issues, burnup credit is still under debate for the use in the waste package design. The probabilistic assessment made by P. Gottlieb (TRW, Las Vegas) and J. Massari (Framatome Cogema Fuels [FCF], Las Vegas) is based on probabilistic pitting corrosion of containers. The pitting corrosion introduces water for the criticality. R. Pope and his associates of Oak Ridge National Laboratory presented the use of depleted-uranium as backfill. Depleted-uranium is expected to dilute fissile materials in the repository. The attendants discussed uncertainties associated with various design options.

In the waste form behavior, spent fuel dissolution, unsaturated tests, and glass leaching were reviewed. R. Stout of Lawrence Livermore National Laboratory (LLNL) presented new models on spent fuel dissolution; K. McCoy of FCF presented oxidation of spent fuel. The dissolution model was too complicated to be used in performance assessment codes and the oxidation model was not new. In the waste package behavior, intermediate results of screening corrosion tests were presented: (a) container materials (A. Roy of FCF and D. McCright of LLNL), and (b) basket materials (R. Van Konynenburg and P. Curtis of LLNL). These short-term tests are not considered valid for the prediction of long-term behavior. Carbon steels in the German rock salt repository (E. Smailos, Forschungszentrum Karlsruhe, Germany) showed only nonuniform corrosion without localized pits. Critical relative humidity (RH) for aqueous corrosion was found to depend on temperature and salt deposits (J. Estill and G. Gdowski of LLNL). This experimental result showed no fixed RH for the initiation of aqueous corrosion, contrary to the models used in Total System Performance Assessment (TSPA)-95.

Dose and biosphere modeling in various nations (Spain, UK, Switzerland, and Canada) was reviewed. Scenarios, critical group, and radionuclide transport were primarily discussed. None of these models consider actinide releases. I discussed colloid-assisted actinide releases with various people in the conference. The importance of colloids is realized, although proper treatments have not yet been made. In the near-field performance assessment, source term with precipitations (T. Ohi, et al., Power Reactor and Nuclear Fuel Development Corporation, Japan) and capillary barriers (M. Apted, et al., of QuantiSci, Inc., S. Tam, et al., of Argonne National Laboratory, and K. Lee, et al., of the Korean Advanced Institute of Science and Technology) were discussed. In the TSPA analyses, Management and Operating Contractor summarized results of TSPA-95. Highlighted topics in the summary include sensitivity of parameters involved, abstraction procedures, and calculated results. Currently, the Division of Waste Management is preparing written review comments of TSPA-95.

CONCLUSIONS AND RECOMMENDATIONS:

The conference was informative and interaction with attendants was useful. The conference agenda and abstracts of highlighted papers are attached for further information. In the future, this conference will be held biannually because of the HLW budget cut, therefore, the participation in the next conference would be more valuable.

Attachments: As stated

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Attachments

1996 INTERNATIONAL HIGH-LEVEL RADIOACTIVE WASTE MANAGEMENT CONFERENCE
APRIL 29 - MAY 2, 1996 · THE MIRAGE HOTEL · LAS VEGAS, NEVADA



	Monday, April 29		Tuesday, April 30	
Room	AM	PM	AM	PM
Mirage Ballroom I/J	8:30 a.m. Opening Plenary			
	11:00 a.m. Plenary--The Role of International and National Regulations Assuring Safe Transport of Radioactive Materials		10:30 a.m. Plenary--Prospects and Perspectives on Nuclear Waste Management	
Mirage Ballroom K/L		1:30 p.m. Total-System Performance Assessment Issues	8:00 a.m. Issues Important to Licensing	1:30 p.m. Regulatory Compliance at the WIPP
		3:30 p.m. Modeling of Groundwater Flow at Yucca Mountain		
Grand Ballroom E		1:30 p.m. Criticality Issues-I	8:00 a.m. Advanced Technologies and Engineering Concepts	1:30 p.m. High Level Waste Management Issues
		3:30 p.m. Criticality Issues-II		
Montego A		1:30 p.m. Microbial Activity Relevant to Nuclear Waste Disposal	8:00 a.m. Probabilistic Volcanic Hazard Analysis Project	1:30 p.m. Politics, Policy and Perceptions of the Yucca Mountain Project--Panel
		3:30 p.m. Microbiological Research at Yucca Mountain		
St. Martin		1:30 p.m. Addressing Environmental Justice Issues in Waste Management Activities	8:00 a.m. Quality Assurance	1:30 p.m. Site Selection and Evaluation in Countries of Eastern Europe
		3:30 p.m. Transportation Issues		
Montego B		1:30 p.m. Deep Geological Disposal Programs in Preparation and Under Development	8:00 a.m. Calibration and Validation of Hydrogeologic Models	1:30 p.m. Waste Form Behavior
		3:30 p.m. Geochemical Studies		
Mirage Ballroom M/N				4:30 p.m. POSTER SESSION--Nuclear Waste Management Topics
				NATURAL
				INTEGRATED
				ENGINEERED
				INSTITUTIONAL



**1996 INTERNATIONAL HIGH-LEVEL RADIOACTIVE WASTE MANAGEMENT CONFERENCE
APRIL 29 - MAY 2, 1996 · THE MIRAGE HOTEL · LAS VEGAS, NEVADA**

	Wednesday, May 1		Thursday, May 2
Room	AM	PM	AM
Mirage Ballroom I/J	10:30 a.m. Finality - Factoring into the Biosphere - The Final Link in Dose Assessment		
Mirage Ballroom K/L	11:30 a.m. Dose and Biosphere Modeling	1:30 p.m. Addressing Technical Uncertainty	8:00 a.m. Yucca Mountain SRA Analysis
		3:30 p.m. Near-Field Performance Assessment	10:00 a.m. Demonstrating Compliance
Grand Ballroom E	8:00 a.m. Rock Mechanics	1:30 p.m. Repository Ventilation	8:00 a.m. Package Thermal Evaluations
		3:30 p.m. Repository Design Issues	10:00 a.m. Waste Package Behavior
Montego A	8:00 a.m. International Perspectives on High-Level Waste Disposal		
		8:30 p.m. Site-Specific Advisory Board - Panel Discussion	
St. Martin	8:00 a.m. Hydrogeology - General	1:30 p.m. Hydrogeological Problems	
		3:30 p.m. Thermal Effects	
Mirage Ballroom M/A			

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IMPACTS OF SNF BURNUP CREDIT ON THE SHIPMENT CAPABILITY OF THE GA-4 CASK

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ABSTRACT

Scoping analyses were performed to determine the impacts of two different levels of burnup credit and two different spent fuel pickup rates on the shipment capability and the minimum fleet size of the GA-4 cask. The analyses involved developing loading curves for the GA-4 cask based on the actinide-only and principal-isotope burnup credit considerations. The analyses also involved examination of the spent nuclear fuel assembly population at nine reactor sites and categorization of the assemblies in accordance with the loading restrictions imposed. The results revealed that for the nine sites considered, depending on the level of burnup credit and the pickup rate assumed, the total savings in shipment and cask fleet costs (1994 dollars) can range from \$55 million to \$74 million.

I. INTRODUCTION

In August 1994, General Atomics (GA) submitted the Safety Analysis Report for Packaging (SARP)⁽¹⁾ to the Nuclear Regulatory Commission (NRC) for the GA-4 cask. The SARP is based on the unirradiated (fresh) fuel assumption or no-burnup credit. This assumption finds the cask capacity to be limited by the initial enrichment (IE) weight percentages for U-235. As a result of this limitation, the allowed capacity of the cask varies from 2 to 4 pressurized water reactor (PWR) spent nuclear fuel (SNF) assemblies.

Burnup credit accounts for the reduction in k -eff (a measure of neutron multiplication) of irradiated fuel (spent nuclear fuel) as compared to the unirradiated fuel when performing criticality control evaluation analyses. This reduction is due to net reduction in fissile material during reactor operation and the buildup of neutron absorbers. Burnup credit falls into three categories: 1) net fissile depletion, 2) net fissile depletion + key actinide absorbers,

and 3) net fissile depletion + key actinide absorbers + key fission product absorbers. The analyses reported in this paper involved categories 2 and 3, referred to as the actinide-only (AO) burnup credit and principal-isotope (PI) burnup credit, respectively.

Taking credit for SNF burnup relaxes the cask-loading restriction imposed by the fresh fuel assumption. By taking credit for burnup, as compared with the no-burnup credit case, the number of four-assembly shipments increases and the number of three-assembly and two-assembly shipments decreases. This introduces a considerable savings in the total number of shipments and, hence, in the associated costs.

When burnup credit is considered, the loading of the GA-4 cask is governed by the cask burnup credit loading curve. The loading curve is developed for a particular cask based on extensive criticality safety analyses and incorporating the SNF characteristics (dimensions, age, burnup, IE); the level of burnup credit; and the capacity of the cask. A loading curve segregates, on the burnup-IE plane, the SNF assemblies into acceptable and unacceptable groups for loading. This segregation of assemblies is based on the IE and minimum burnup of the SNF assemblies.

This paper reports the results of scoping analyses performed to determine the impact of SNF burnup credit on the number of shipments and the minimum fleet size of the GA-4 cask.

II. WORK DESCRIPTION

The scoping analyses conducted assumed nine truck-onl PWR sites as a reasonably conservative assumption: Crystal River 3, Fort Calhoun, Ginna, Haddam Neck, Indian Point Indian Point 2, Indian Point 3, St. Lucie 1, and Yankee Row

U.S. LIGHT-WATER REACTOR SPENT FUEL INVENTORY—FISSILE DISTRIBUTION

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ABSTRACT

A study was performed to characterize the fissile distribution of the U.S. light-water reactor (LWR) spent fuel inventory that has been discharged from U.S. commercial reactors through December 31, 1993. Data are presented which give the total mass of discharged LWR spent fuel in metric tons of initial heavy metal as a function of the final effective enrichment of the discharged LWR spent fuel.

I. INTRODUCTION

Those conducting waste management studies¹ to reduce the potential for nuclear criticality in a future geological repository must examine the quantities and distribution of fissile isotopes that are present in discharged boiling-water reactor (BWR) and pressurized-water reactor (PWR) spent nuclear fuel (SNF) scheduled for disposition. The major fissile isotopes present in LWR fuels that impact criticality safety are the nuclides, ^{235}U , ^{239}Pu , and ^{241}Pu . The sum of the quantities of these three nuclides, expressed as a percentage of the total amount of all U and Pu isotopes present in a batch of discharged fuel, determines the final effective enrichment of the fuel batch under consideration. The final effective enrichment provides an approximate measure of the nuclear criticality potential. As the final effective enrichment increases, the mass, geometry, or administrative controls that must be in place to prevent nuclear criticality become more stringent. Below an enrichment of about 0.7%, however, criticality is no longer a concern because the infinite multiplication factor for any heterogeneous or homogeneous mixture of fuel and water, even under conditions of optimum moderation, is less than unity. The current study examines the distribution of the final effective enrichment of the LWR SNF which was discharged through December 31, 1993, and which currently resides in the fuel storage pools of the various utilities or in one of several AFR facilities.

II. ASSUMPTIONS

The information on which this study is based is the SNF data contained in Nuclear Fuel Data Form² RW-859. Historical inventories of discharged LWR SNF recorded on this data tape have been updated through December 31, 1993. The total inventory of BWR fuel discharged from U.S. commercial reactors through December 31, 1993, amounts to 10,178.53 metric tons of initial heavy metal (MTIHM), while that for discharged PWR fuel amounts to 17,851.69 MTIHM. The data include the quantity (MTIHM) of fuel contained in each discharge batch, the initial enrichment of the batch, and the final burnup at discharge for each batch (expressed in MWd/MTIHM).

III. METHODOLOGY

The methodology used to determine the distribution of the final effective enrichment of the historical inventory of discharged LWR SNF began with the creation of a 3-D array whose members represent the quantities of discharged LWR fuel contained in the various discharged batches, categorized by (a) fuel type, (b) initial enrichment, and (c) burnup. The range of initial enrichment (0–5%) was divided into 50 equal increments of 0.1% each; the range of fuel burnup (0–60,000 MWd/MTIHM) was divided into 24 equal increments of 2,500 MWd/MTIHM each. Initial enrichment and burnup values at the midpoint of each increment were assigned to each of the array members to determine the final effective enrichment at discharge.

Next, the ORGENTRE (Oak Ridge Waste GENERation and TREATment) Code, a series of computer codes and data bases which characterize nuclear waste generation and treatment, was run to calculate the nuclide composition for BWR and PWR fuel corresponding to each combination of initial enrichment and final burnup increment midpoint values. The ORGENTRE code³ is based on the ORIGEN2 isotope generation and decay code,⁴ which

PROBABILISTIC EVALUATION OF POSTCLOSURE CRITICALITY EVENTS INTERNAL TO THE WASTE PACKAGE

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INTRODUCTION

The regulations governing disposal criticality require that criticality not be possible, except under certain very unlikely sequences of events, and that the risk of criticality (probability times consequences) not violate repository performance guidelines. This paper describes the analysis and results of a probabilistic evaluation of the likelihood that the events/conditions required for a criticality event internal to the waste package (WP) during the postclosure phase of the repository will occur^{1,2}. This analysis supports a determination of licensability as it is affected by criticality control.

WORK DESCRIPTION

For a loaded and sealed WP, several events or conditions are required before a criticality event is possible. These are:

1. A source of water with access to the waste package,
2. Breach of all WP barriers, allowing moderator entry to WP interior,
3. Loss of neutron absorber from the basket structure,
4. Fuel geometry that provides for sufficient moderation/reflection,
5. A WP which is capable of holding water, and
6. Fuel that is isotopically capable of criticality given a source of moderator, an appropriate geometry, and insufficient neutron absorbers.

During the preclosure phase of the repository, the likelihood of a criticality is generally characterized by the independent occurrence of the above events. Sequences of dependent events (i.e., corrosion) are extremely unlikely due to the relatively small amount of time elapsed in the preclosure phase. However, during postclosure, the likelihood of dependent sequences of events increases as more time passes, while some of the possible independent causes of these events which existed during preclosure are no longer present. Event/condition 6 is independent of the other four events in terms of the fuel isotopics, while the fuel cladding is relatively intact.

For this analysis, only the dependent occurrence of events 1 through 3 were evaluated. The basket and fuel geometries, event/condition 4, were assumed to remain intact. Three dependent sequences of events were defined; initiation (event 1) by water dripping on the WP at a low rate (1a), initiation by water dripping at a high rate (1b), and flooding (1c). The time to occurrence of events 2 and 3 is conditional on the occurrence and magnitude of event 1, because an aqueous environment is required to drive the corrosion processes. The probability density functions (PDFs) for the time to occurrence of each of the three events in the three sequences were estimated from available data on corrosion and infiltration processes.

Next, the PDF for the time to occurrence of a specific sequence of dependent events is given by the convolution of the PDFs of the three events in that sequence. For example, if f_1 is the PDF for an initiating environmental condition, f_2 is the PDF for a WP barrier breach, which is conditional on f_1 , and f_3 is the PDF for neutron absorber removal, which is conditional on f_2 , then the PDF for the time to occurrence of all three failures is mathematically

DEPLETED-URANIUM-SILICATE BACKFILL OF SPENT-FUEL WASTE PACKAGES FOR REPOSITORY CONTAINMENT AND CRITICALITY CONTROL

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ABSTRACT

A new technology,¹ the Depleted Uranium Silicate Container Backfill System (DUSCOBS), is proposed to improve the performance and reduce the uncertainties of geological disposal of spent nuclear fuel (SNF). DUSCOBS is designed to: (1) reduce radionuclide release rates from the waste package and (2) reduce the potential for repository nuclear criticality events. As a secondary benefit, it disposes of excess depleted uranium (DU). DUSCOBS also provides benefits for SNF storage and transport. These are discussed in a companion paper.²

I. MECHANICAL AND CHEMICAL DESCRIPTION

SNF repository waste packages are filled with SNF and then backfilled with depleted uranium (0.2 wt % ²³⁵U) silicate (DUS) glass beads, which are sufficiently small (<1 mm) to fill coolant channels in the SNF assemblies. The glass is in excess of 20 wt % uranium with DU sufficient to lower the total fissile concentration in the waste package below 1 wt % heavy metal. A high performance glass is chosen with dissolution kinetics similar to SNF. The thermal properties of the waste package with and without DUSCOBS are similar.²

II. REPOSITORY BENEFITS

A. Reduction of Repository Radionuclide Release Rate

DUS glass lowers the long-term radionuclide release rate from the waste package. Most fission products and actinides in SNF are incorporated into uranium dioxide (UO₂) pellets. These fission products and actinides can not escape until the UO₂ dissolves or is transformed into other chemical species. The DUS glass saturates the groundwater in the waste package with uranium and slows the SNF dissolution process. Uranium-saturated groundwater can not dissolve added SNF uranium.

The silicate in the glass lowers the solubility of uranium in groundwater; thus, more water must flow through a waste package to remove a unit quantity of uranium. This delays the dissolution of the SNF UO₂. Uranium silicate also tends to form a coating on the UO₂ which acts as a further barrier to the dissolution and transformation of UO₂.

B. Reduced Potential For Repository Nuclear Criticality

DUSCOBS reduces the potential for both package and zonal repository nuclear criticality events in a repository. In a repository, long-term, low-power nuclear criticality events are a major concern because they generate heat that (1) accelerates degradation of the waste packages and (2) accelerates water movement that can transport radionuclides to the environment.

PITTING CORROSION OF CONTAINER MATERIALS IN ANTICIPATED REPOSITORY ENVIRONMENTS

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ABSTRACT

Electrochemical corrosion tests were performed at the Lawrence Livermore National Laboratory to evaluate the susceptibilities of candidate waste package container materials to pitting corrosion in environments relevant to the potential repository. The results obtained from these tests will be used in identifying a group of prospective container materials having the desired corrosion resistance.

1. INTRODUCTION

Metal-barrier waste package design concepts consisting of an outer corrosion-allowance metallic barrier over a corrosion-resistant metallic inner container are currently being considered to accommodate the nation's spent nuclear fuel and vitrified high-level nuclear waste for geologic disposal. Although the groundwater in the proximity of the potential repository is believed to be non-aggressive for many corrosion-resistant metallic materials, under some operating conditions, the repository environments surrounding these waste packages can become very hostile. Under these conditions, the corrosion-resistant metallic container may become susceptible to localized corrosion, such as, pitting. This paper presents the results of pitting corrosion studies of candidate container materials in three different aqueous environments incorporating an electrochemical test technique.

II. WORK DESCRIPTION

Materials tested include austenitic Types 304 and 316L stainless steels, nickel-rich Alloys 825 and G-3, nickel-base Alloys C-4 and C-22, titanium-base Ti Gr-12, nickel-copper Alloy 400, and 70/30 Cupronickel. Tests were performed in deaerated neutral, acidic, and alkaline salt solutions (52.6 grams of NaCl per liter of solution) at room temperature, 60°C, and 90°C. The pH of these

solutions ranged between 6 and 7, 2 and 3, and 10 and 11, respectively.

A three-electrode technique was used to conduct cyclic potentiodynamic polarization (CPP) experiments using a cylindrical working electrode, two graphite counter electrodes, and a Luggin capillary connected to a reference electrode. While a saturated calomel reference electrode (SCE) was used at room temperature and 60°C, Ag/AgCl was used at 90°C as the reference electrode. Potential was applied to the specimen at the ASTM-specified⁽¹⁾ rate of 0.17 mV/sec using a Model 273 potentiostat, controlled by an IBM-compatible PC with Model 252/352 Softcorr II software, both manufactured by EG&G Instruments, Princeton Applied Research. A controlled temperature water bath maintained the test temperature at the desired value within $\pm 0.1^\circ\text{C}$.

Initially the specimen was allowed to reach its stable corrosion potential (E_{corr}), followed by an anodic potential scan. The changes in current were automatically recorded, and the resulting polarization curve was displayed on the computer monitor. After polarization to +200 mV (versus SCE or Ag/AgCl), the direction of scanning was reversed at the same rate to E_{corr} , and the CPP plot was printed. The pH of the solution was measured before and after the test. The tested specimen was cleaned, followed by visual and microscopic evaluations.

III. RESULTS AND DISCUSSION

Results indicate that of all the materials tested, Alloy C-4, Alloy C-22, and Ti Gr-12 demonstrated the maximum corrosion resistance, showing no pitting or dissolution in any environment. However, Alloys 825 and G-3 suffered from pitting in acidic concentrated brine at 90°C. Both Types 304 and 316L stainless steels showed severe pitting tendency in all three environments at 60°C and 90°C. Furthermore, Alloy 400 and 70/30 Cupronickel were readily susceptible to dissolution under all

WATER VAPOR EFFECTS ON THE CORROSION OF STEEL

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I. INTRODUCTION

These studies were conducted in support of the Yucca Mountain Site Characterization Project (YMSCP) and are part of the characterization of the behavior of metallic materials in expected environments at a potential high-level radioactive waste repository at Yucca Mountain, Nevada. The potential repository is above the water table, and waste packages will be exposed to elevated temperatures and potentially high water partial pressures. A proposed outer barrier material, in a multi-barrier containment concept, includes carbon steel. Inner barrier materials would be constructed of a more corrosion resistant material such as a titanium based alloy, Ni-Cr-Mo or Ni-Fe-Cr-Mo alloy. The focus of this work is to determine the corrosion/oxidation performance of the outer barrier material.

For this study, the effects of water vapor partial pressure on the corrosion/oxidation behavior of carbon steel in the temperature range of 50 - 110 °C at atmospheric pressure were investigated using thermogravimetric analysis. At temperatures above the boiling point, which is dependent on the presence of hygroscopic salts and surface condition, the interest is in the effect of water vapor on dry oxidation. At temperatures below the boiling point, where water condensation may exist, the interest is in the presence of thin water films in which electrochemical processes may occur. The presence and corrosivity of thin water films are dependent on temperature, water partial pressure, surface condition, gaseous constituents, and adsorbed hygroscopic salts.

II. EXPERIMENTAL

The experiments were conducted in a modified thermogravimetric analyzer (TGA) which continuously measures the weight change of a specimen while it is undergoing oxidation or corrosion processes in the presence of various partial pressures of water. The weight measurement resolution of the TGA is approximately 60 µg.

The temperature of the reaction chamber is controlled by circulating fluid through an annular chamber formed by

a double-walled glass cylinder. The fluid, which is continuously circulated in a closed loop between the annular chamber and a temperature-controlled bath, provides a temperature deviation of less than 0.5 °C along the length of the specimen contained inside the inner wall of the glass cylinder (reaction chamber).

The relative humidity (RH) is varied in the reaction chamber by mixing precise amounts of vaporized water and dry air. The vaporized water is provided by a vaporizer/micrometering pump assembly, and the dry air is supplied using a mass flow controller. Controlled mixing of the vaporized water with the dry air is the means used to vary the relative humidity.

Test specimens measured 3.8 cm x 1.3 cm x 0.16 cm. The nominal weight of a specimen is 5.6 grams. The specimens were suitably degreased with isopropanol prior to testing.

The test parameters of time, temperature, weight, and relative humidity were recorded and stored electronically.

III. RESULTS

Preliminary results for AISI 1020 carbon steel specimens exposed to a range of humidified environments at 65 °C were consistent with aggressive oxidation at ≥ 85% RH and little oxidation at and below 75% RH (see Figure 1). The aggressive electrochemical corrosion processes at or above 85% RH exhibit rapid corrosion initially but slow dramatically after the first 24 hours.

These results indicate that in the range of 75-85 % RH the transition from dry oxidation to electrochemical corrosion is occurring. The critical relative humidity value in this range is similar to that observed for carbon steels at ambient temperatures.¹

Visual examination of those specimens exposed to 85% RH showed uniform reddish-brown corrosion product present on the surface, while those specimens exposed to lower relative humidity were virtually non-oxidized. Examination conducted with an optical microscope at

IMPACTS OF CATHODIC PROTECTION ON WASTE PACKAGE PERFORMANCE

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I. INTRODUCTION

The current design concept for a multi-barrier waste container for the potential repository at Yucca Mountain, Nevada, calls for an outer barrier of 100 mm thick corrosion-allowance material (CAM) (carbon steel) and an inner barrier of 20 mm thick corrosion-resistant material (CRM) (Alloy 825). Fulfillment of the NRC subsystem requirements (10 CFR 60.113) of substantially complete containment and controlled release of radionuclides from the engineered barrier system (EBS) will rely mostly upon the robust waste container design, among other EBS components. In the current waste container design, some degree of cathodic protection of CRM will be provided by CAM. This paper discusses a sensitivity case study for the impacts of cathodic protection of the inner barrier by the outer barrier on the performance of waste package.

II. WORK DESCRIPTION

Incorporating humid-air and aqueous general and pitting corrosion models (including their uncertainties) of CAM^{1,2} and the pitting corrosion model (including its uncertainty) of CRM³, a detailed stochastic waste package performance simulation model was developed for the 1995 total system performance assessment (TSPA)⁴. Currently, no data or adequate models for cathodic protection (or galvanic corrosion) between two metals are available, which may be applicable to waste package degradation simulation. An expert elicitation⁵ was provided to account for the cathodic protection of the corrosion-resistant inner barrier in the waste package. The elicitation suggests that pitting corrosion of the inner barrier be delayed until the thickness of the CAM is reduced by 75%. The cathodic protection measure was incorporated into the stochastic waste package simulation model.

The near-field conditions considered in this study were for the case with a thermal loading of 83 MTU/acre, without backfill around the waste package and an infiltration rate of 0.3 mm/yr through the repository. A typical waste package for 21 pressurized-water-reactor assemblies has an outside

length of 568.2 cm and an outside diameter of 180.2 cm⁶. Relative humidity and temperature profiles at the waste container surface were modeled using a drift-scale thermal-hydrologic model⁴. The temperature and relative humidity profiles at the waste package surface for the above are shown in Figure 1.

Incorporating the temperature and relative humidity profiles into the stochastic waste package simulation model, simulations were performed for cases without and with cathodic protection. Each case was simulated with a total of 400 waste packages and 250,000 pits per waste package.

III. RESULTS AND DISCUSSION

The simulation results for the case without cathodic protection are presented in Figure 2 which shows a "failure" history (or empirical CDF) of waste packages. "Failure" of waste package is defined as having at least one pit penetration. In the near-field conditions discussed above, the time for the first waste package failure occurs at about 2,000 years, and the number of failed waste packages increases rapidly, reaching about 90 % failures by 10,000 years and about 96 % failures by 100,000 years.

The empirical CDF for waste package failure history for the case with cathodic protection is shown in Figure 3. The time for the first waste package failure occurs at about 8,000 years, providing an additional 6,000 years of waste package performance compared to the case without cathodic protection. Waste package failure rates are also significantly lower than those without cathodic protection. The cumulative fraction of waste packages that have failed by 10,000 years is negligible, and is about 60 % by 100,000 years. Also, shown in the figure is the cumulative fraction of waste packages with the outer barrier thickness reduced by 75%.

IN-SITU CORROSION TESTING OF SELECTED HLW CONTAINER MATERIALS

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I. INTRODUCTION

To qualify corrosion resistant materials for long-lived HLW containers that could act as a barrier for immobilization of radionuclides in a rock salt repository, the corrosion behavior of preselected materials is being examined under simulated disposal conditions.

In the present study, long-term in-situ corrosion experiments (field experiments) were performed in the Asse salt mine on materials which were identified as promising in previous work^{1,2}. These are: the passively corroding alloys Ti 99.8-Pd and Hastelloy C4 (a Cr-Ni-Mo alloy) as reference materials for a corrosion resistant concept, and the actively corroding TStE 355 carbon steel as reference material for a corrosion allowance concept. Besides these materials, the Cr-Ni steel 1.4833 as potential canister material for vitrified HLW was also investigated. The corrosion experiments were performed in the framework of the HLW test disposal in the Asse salt mine under conditions prevailing in the normal operating phase of the repository.

II. DESCRIPTION OF WORK

The materials were investigated in the hot-rolled condition and had the following composition in wt. %:
Ti 99.8-Pd: 0.17 Pd; bal. Ti.
Hastelloy C4: 16.8 Cr; 15.9 Mo; 0.33 Ti; bal. Ni.
Cr-Ni steel (1.4833): 23 Cr; 15 Ni; bal. Fe.
TStE 355 carbon steel: 0.17 C; 0.44 Si; 1.49 Mn; bal. Fe.

Plane specimens (40 mm x 20 mm x 3 mm) of the above-mentioned materials were stored in a 15 m deep cased and electrically heated borehole at the 800 m level

of the Asse salt brine (Fig.1). The specimens were installed on the external wall of the casing (steel coated with Ti 99.8-Pd) so that the creeping salt could come into contact with them. The test temperature was 190°C and the test duration about 5.3 years. During the experiment, the specimens were exposed to water and gases (O₂, CO₂, CH₄, H₂) thermally released from the rock salt into the borehole³. A release of totally 1.1 l H₂O was detected after 5.3 years test duration.

The general corrosion of the materials was calculated from the gravimetrically determined weight losses and the material density. Evaluation of the specimens with regard to local corrosion and stress corrosion cracking was made by microscopic evaluation, measurements of pit depths, surface profilometry and metallography.

III. RESULTS

The corrosion results obtained for the various materials are summarized in Table 1. The materials Ti 99.8-Pd and Hastelloy C4 corroded, as expected, at extremely small general corrosion rates (<0.1 µm/a). Furthermore, Ti 99.8-Pd was completely resistant to local corrosion such as pitting or crevice corrosion. In case of Hastelloy C4, however, some small pits of about 10 µm depth were observed after 5.3 years testing in the crevice formed between Hastelloy specimen and borehole salt wall. General corrosion of the Cr-Ni steel 1.4833 was very low (about 1 µm/a), but the material exhibited severe pitting corrosion (up to 200 µm after 5.3 years) and stress corrosion cracking. Finally, the TStE 355 carbon steel was resistant to pitting and crevice corrosion. This steel shows, as in previous studies⁴, a non-uniform general corrosion

CORROSION TEST ON CANDIDATE WASTE PACKAGE BASKET MATERIALS FOR THE YUCCA MOUNTAIN PROJECT

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ABSTRACT

A scoping corrosion test was performed on candidate waste package basket materials in order to assist in selecting materials for package design and to help in designing longer-term corrosion tests. The corrosion solution was buffered near pH4, was in contact with air, and contained chemical species expected to be produced by radiolysis. The test was conducted at 90°C for 96 hours. Samples included aluminum-, copper-, stainless steel-, and zirconium-based metallic materials and several ceramics, incorporating neutron absorber elements. Sample weight losses and solution chemical changes were measured. Both corrosion of the host materials and dissolution of the neutron absorber elements were studied.

I. INTRODUCTION

Waste package designs for spent fuel incorporate basket assemblies to provide structural support for the fuel assemblies, to assist in conduction of the decay heat, and to ensure control of nuclear criticality if liquid water should enter breached packages. The entry of substantial amounts of liquid water into the packages is considered unlikely, given expected conditions at the Yucca Mountain site and the designs of the repository and the waste packages. Nevertheless, it is prudent to prepare for this eventuality, particularly in view of the very long half-lives of the fissile species, and in view of current uncertainties in the final regulatory requirements for criticality safety in a repository.

It is not possible to predict accurately when a particular waste package would fail, what the temperature would be at that time, and when liquid water might be available to enter. It is also therefore not possible to accurately predict what the radiation dose rate would be at the time these events occurred. Since all these factors are important in determining the water chemistry in a breached package, and since the ensemble of packages would be subject to a range of external conditions, it is not possible to specify an exact composition of the corrosion environment to which the basket materials would be subjected.

In this situation, we have chosen to perform a scoping corrosion test that simulates, by ordinary chemical means, the effects of irradiation, recognizing that transient ionic and free radical effects produced by irradiation are thereby not exactly reproduced. We have used a solution that contains a significant concentration of chloride and greater than expected concentrations of the stable species that are known to be produced in the radiolysis of air-water systems. Chloride is found in vadose water and is known from past experience to have profound effects on the corrosion resistance of many materials. The relatively high concentrations of radiation-produced species were used to increase the sensitivity of the test, since its duration was necessarily orders of magnitude shorter than the times of interest for disposal of spent fuel and high level waste. We have chosen to test at a temperature near the high end of the range possible for liquid water at the expected atmospheric pressure in the potential repository. These

MICROBIAL STUDIES IN THE CANADIAN NUCLEAR
FUEL WASTE MANAGEMENT PROGRAM

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The Canadian concept for nuclear fuel waste disposal is based on a multiple barrier system. It involves disposing of used CANDU (CANada Deuterium Uranium) fuel, contained in corrosion-resistant containers (titanium or copper) surrounded by compacted clay-based buffer and backfill materials, deep in a vault excavated in a stable granitic formation.

An accelerated microbial program has been carried out at AECL since 1991 to study the potential effects of microbial activity on the integrity of the proposed multi-barrier system. The current approach is focussed on answering specific questions, in areas such as the survival of microbes in buffer materials under relevant radiation, temperature and moisture conditions; the potential for microbially-influenced corrosion (MIC) of container materials; transport of microbes through highly compacted buffer material in relation to MIC and radionuclide migration; the introduction of nutrients through human presence and activity during vault construction and operation; the potential for microbial gas production from organics naturally present in buffer

and backfill materials; the presence and activity of microbes in deep granitic groundwaters and the effects of biofilms on radionuclide migration in the geosphere.¹

Several studies at AECL have shown that doses of radiation, combinations of heat and radiation, and desiccation (as a result of heat), similar to what would be expected in a vault, each have detrimental effects on the natural viable bacterial populations in buffer materials. In a full-scale test, carried out at AECL's Underground Research Laboratory (URL), a simulated waste container (maximum heat output 85°C) was buried for 2.5 years, surrounded by compacted buffer material. Extensive microbial analysis of this experiment has shown that most viable organisms in buffer material disappear around a moisture content of 15% (corresponding to a water activity (a_w) of between 0.95 and 0.96).² Results from radiation experiments were used to deduce that the surface of nuclear fuel waste containers would be 'sterilized' in 9 to 33 d after emplacement, depending on the type of container, the age of the fuel and the initial concentration of microbes.³

MICROBIALY-INFLUENCED CORROSION CAPABILITY OF YUCCA MOUNTAIN BACTERIAL ISOLATES

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I. ABSTRACT

Microorganisms implicated in microbially-influenced corrosion have been isolated from the deep subsurface at Yucca Mt. Iron-oxidizing (FeOx), sulfate-reducing (SRB), and exopolymer (EPS)-producing bacteria were found. Microbial corrosion rate was monitored electrochemically. The test system was composed of a 1020 carbon steel coupon immersed in soft R2A agar prepared with simulated groundwater (J-13). A KCl bridge was used to connect the test and reference cell (calomel electrode). A platinum counter-electrode was used to apply a potential to the coupon and the corrosion process was measured by a potentiostat (Gamry). Corrosion cells (3x) were inoculated with purified cultures of EPS-producing bacteria and enrichment cultures of FeOx and SRB bacteria. Test cells were inoculated with microorganisms separately, as well as in various combinations. An uninoculated control cell was prepared to assess abiotic corrosion. Average corrosion rates were measured in milli-inches per year (mpy) against time. The control, and cells containing EPS-producing, FeOx or SRB bacteria alone or in combination demonstrated a rapid decrease in corrosion rate by 3 days. The corrosion rates stabilized, and at 35 days peaked at 2.25 mpy (FeOx), 3.30 mpy (SRB), and 2.80 mpy (EPS). All of these values were significantly higher than the corrosion rate observed in the control cell, 1.30 mpy at 35 days. The various combinations demonstrated higher corrosion rates than any bacterial group alone. Coupons were cleaned, revealing surface pits. 200 pits/sq. in. were counted on a coupon previously exposed to a mixture of EPS-producing and FeOx microorganisms. Pit diameter ranged from 0.25 to 2.75 mm. The results indicate that Yucca Mountain microorganisms, alone and in combination, are capable of causing corrosion of 1020 carbon steel.

II. INTRODUCTION

Corrosion is traditionally regarded as a sequence of electrochemical reactions at a metal surface in contact with an aqueous electrolyte-containing solution (Geesey, 1990). This results in the dissolution of metal from anodic sites with subsequent electron acceptance at cathodic sites (Ford et al., 1990). It is now known that corrosion reactions may actually be induced or enhanced by microbial activity (Ford, et al., 1990). Historically, microorganisms implicated in microbially-influenced corrosion include iron-oxidizing, sulfate-reducing, acid-producing, and exopolymer-producing bacterial populations. The involvement of these bacteria in corrosion is a testament to their ability to adapt to favorable circumstances in man-made systems analogous to their own ecological niches (Costerton, *et al.*, 1990). When a metallic surface is immersed in an aqueous solution a conditioning film forms, attracting bacteria, algae, and other microorganisms to the surface, initiating biofilm formation (Borenstein, 1993). In natural environments, microorganisms can exist alone but are more commonly found in complex consortia within a biofilm. The biofilm contributes to the formation of an oxygen gradient, creating anaerobic zones in the portion of the biofilm in contact with the metal surface. This provides ideal conditions for the development of corrosion processes. The result is metal dissolution and pit formation (Borenstein, 1993). The microorganisms utilized in this research project were previously isolated from rock samples obtained from the proposed nuclear waste repository site at Yucca Mountain. It is imperative to understand what effects microbial metabolism may have on materials chosen for site construction and waste containment. Loss of integrity of the metal canisters could result in leakage of radioactive waste into the surrounding environment. One of the metals being considered for the outer barrier

URANIUM DIOXIDE DISSOLUTION UNDER ACIDIC AQUEOUS CONDITIONS

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I. INTRODUCTION

Understanding of the long-term dissolution of waste forms in groundwater is required for the safe disposal of high level nuclear waste in a geologic repository, because waste-form radionuclides could be released by dissolution and transported in groundwater. The dissolution of the uranium dioxide (UO_2) matrix in spent nuclear fuel is considered the rate-limiting step for release of radioactive fission products. The intrinsic UO_2 dissolution rate sets an upper limit on the aqueous radionuclide release rate.

Unsaturated spent fuel tests¹ have shown that pH's of leachates have decreased to a range of 4 to 6, presumably due to air radiolysis that oxidizes nitrogen, producing nitric acid. Dissolution rates under such acidic conditions may be different than those previously reported for alkaline groundwater conditions. No dissolution rate measurements of UO_2 or spent fuel have been reported for acidic conditions possibly relevant to a geologic repository.

The purpose of our work has been to measure the intrinsic dissolution rates of uranium dioxide under acidic conditions that are relevant to a repository and allow for modeling. Experiments have been completed at room-temperature and 75°C, pH's of 4 and 6, and air and oxygen saturated aqueous solutions. These are compared with earlier work on spent fuel and UO_2 using alkaline solutions.²

II. DESCRIPTION

A planned set of 27 UO_2 dissolution experiments was developed using statistical experimental design methodology. These experiments would allow a systematic evaluation of the effects of temperature (25-75°C), pH (4-6) and dissolved oxygen (80-8000 ppb) concentrations on the uranium dissolution rate. The initial results reported here include dissolution rates for two UO_2 polycrystalline samples at 25 and 75°C, and pH's 4 and 6, all at atmospheric oxygen conditions, 8 ppm dissolved oxygen.

The experiments use single-pass flowthrough conditions to prevent precipitation and other competing reactions from distorting dissolved uranium concentration measurements. The samples were from the same batch of polycrystalline UO_2 used in previous experiments.⁽²⁾ Nitric acid was used to adjust pH.

III. RESULTS AND CONCLUSIONS

Figure 1 shows a time history of the dissolution results. The sample 1 (S1) leaching solution was in at a pH of 3.8 (4), while the sample 2 (S2) leaching solution initially had a pH of 5.9 (6). Both samples initially at room temperature (~20°C). The dissolution rates (DR) appear similar in the range of 2 to 6 mg/(m²-day). After day 11 the buffer solutions were switched. Sample 2, now at pH=4, initially increased dissolution rate and sample 2, switched to pH=6, decreased. They both returned closer to their earlier values although the ratio of dissolution rates [DR(pH=4)/DR(pH=6)] seems to have increased. At day 20 the leaching solutions were switched back to original samples and the temperature increased to 75°C. The dissolution rate of sample 1, again at pH=4, increased about ten-fold while sample 2 at pH=6 unexpectedly remained about the same. After four more samplings the sparge gas was switched to air, with its small fraction of CO_2 . This did not have any readily apparent effect on the dissolution rates. The samples were returned to room temperature after day 30. The pH=4 dissolution rate returned to near its original room-temperature value while the pH=6 sample rate dropped below its previous room temperature amount to a barely detectable level.

Previously measured room-temperature UO_2 dissolution rates in alkaline, low-carbonate waters were 3.9 mg/(m²-day) at a leaching solution pH of 8 and 2.6 mg/(m²-day) at a pH of 4. In the pH ranges of 4 to 6 and 8 to 10, these early results indicate that there is no significant effect of acidity or alkalinity on room-temperature UO_2 dissolutions. The ten-fold effect on dissolution rate of increasing temperature of the pH=4 sample seems closer to the

RADIONUCLIDE RELEASE FOR UNSATURATED SPENT FUEL TESTS - FIRST 1.6 YEARS

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I. INTRODUCTION

To provide input to evaluate the reaction of spent fuel for unsaturated conditions that may be expected in the Yucca Mountain candidate repository, we have ongoing tests at 90°C in which a simulated groundwater, EJ-13, is dripped at 0.75 mL at 3.5 day intervals on spent fuel samples. This paper examines the release behavior of ^{238}U , ^{137}Cs , ^{90}Sr , ^{99}Tc , and ^{129}I to determine if any of them can represent matrix dissolution of spent fuel. The dissolution results reported from flow-through tests^{1,2} are used to provide a measure of the potential magnitude of matrix dissolution.

II. DESCRIPTION

Two pressurized-water-reactor fuels are used: ATM-103, fission gas release of 0.25%, and ATM-106, fission gas release of 11.2%. The experimental configuration and the composition of the leachant, EJ-13, is described elsewhere³. Cation content in aliquots of the leachate and the acid solutions used to strip the test vessel was determined using inductively coupled plasma-mass spectrometry. Details are given elsewhere⁴. The ^{137}Cs content was obtained from gamma spectrometry data.

III. RESULTS AND DISCUSSION

The location of ^{137}Cs , ^{90}Sr , ^{129}I , and ^{99}Tc in a spent fuel pellet as reported by Gray et al¹ and Wuertz and Willinger⁵ is summarized in Table 1. Because the fuels used in these tests are the same

as Gray's, we will use his ^{99}Tc values for comparison purposes. Wuertz's values for ^{99}Tc may not be representative for our fuels since ATM-103 contains large amounts of Xe particles along the grain boundaries, which suggests that ^{129}I is located there, and ATM-106 has a large fission gas release (11.2%), which suggests a large ^{129}I release to the gap.

The fractional releases and corresponding daily release rates for ^{238}U , ^{137}Cs , ^{90}Sr , ^{129}I , and ^{99}Tc are listed in Table 2. The ^{238}U fractional release is smaller than that of any of the other radionuclides. This finding can be correlated with the formation of large amounts of alteration products on spent fuel, including a billietite-like phase containing barium, cesium and a very small amount of strontium. The ideal billietite formula is $\text{Ba}(\text{UO}_2)_6\text{O}_4(\text{OH})_6 \cdot 8\text{H}_2\text{O}$. The incorporation of cesium and strontium in alteration products reduces their measured release. For both fuels, the ^{90}Sr release fractions were comparable to the ^{137}Cs fractions, ranging between 10^{-7} to 10^{-6} /d in the ATM-106 test and falling from an initial value of 2×10^{-4} to 4×10^{-6} /d in the ATM-103 test.

We did not observe an initial large release, which has often been attributed to ^{137}Cs release. Rather, ^{137}Cs release from ATM-103 fuel was relatively constant during the first 1.6 years of reaction while for the ATM-106 fuel, a delay was noted with significant release after four months of reaction. Since the cumulative released fraction in our tests is significantly smaller than the ^{129}I and the

NONEQUILIBRIUM THERMODYNAMICAL MODEL FOR SPENT FUEL DISSOLUTION RATE

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INTRODUCTION

This brief paper discusses work in progress to develop a non-equilibrium thermodynamic model for the dissolution response of uranium oxide spent fuels waste forms. The objective is to derive function forms for the dissolution rate that are consistent with quasi-static, irreversible thermodynamic processes. These function forms contain thermodynamic chemical potentials of both the solid (spent fuels) and the solution (water chemistries) along with a set of coefficients and parameters that can be evaluated by numerical regression of dissolution test data. Currently, detailed knowledge is not available for the atomic (mechanistic) steps and the sequence of chemical/electrochemical reaction steps to describe the dissolution process over the range of spent fuel inventory, potential water chemistries, and temperatures. The existing approach is obtaining an experimental data base of dissolution rates for a subset of spent fuels over a range of controlled, aggressive water chemistries and temperatures. With a numerical regression algorithm, these data are used to evaluate empirical parameters in a rate law.⁹ The function form of this rate law is a product polynomial of the bulk water chemistry concentrations and temperature.¹⁰ In its present form, this function form does not have an explicit thermodynamic dependence on the uranium oxide waste form. In addition, the use of bulk concentrations in the function form for the regression analysis of the dissolution data would not explicitly account for a dependence from possible surface to bulk concentration differences due to surface adsorption and dipole layers. The following thermodynamic model uses analysis methods and physical concepts taken primarily from classical mechanics,^{4,7} colloidal foundations,⁶ thermodynamics,^{3,5,8} electrochemistry,^{1,2} and geochemistry.¹⁰

NONEQUILIBRIUM THERMODYNAMIC DISSOLUTION RATE FUNCTION FORMS

In the following, thermodynamic internal energy functionals are used to represent the energy responses for a generic solid and a generic liquid. The solid and liquid are in contact at an idealized wetted surface. This wetted surface is a material discontinuity, and it is also a dissolution front that propagates at an idealized dissolution velocity, \underline{v} .

The generic solid will have bulk constituents of typical UO_2 spent fuel, namely minor concentrations of other actinides, fission products, and defects in the bulk lattice structure denoted by a column matrix $\{f_S\}$. The bulk lattice is assumed to be nominally that of the UO_2 lattice structure; however, other oxide phases and adsorbed complexes as a result of solid-water chemistry reactions may exist on and in spatial neighborhoods of the wetted surface. These are denoted by a column matrix $\{f_{SL}\}$. The generic liquid will be represented with a subset of arbitrary initial/bulk constituents denoted by column matrix $\{f_L\}$. In addition the liquid has two subsets of water chemistry products formed from the dissolution of the solid and denoted by column matrices $\{f_{LS}\}$ and $\{f_{LSL}\}$. The new reaction product constituents $\{f_{SL}\}$, $\{f_{LS}\}$ and $\{f_{LSL}\}$ are interrelated to the initial solid and liquid constituents $\{f_S, f_L\}$ by stoichiometric chemical reaction expressions.

Each of the constituent densities of the solid and the liquid will be assumed to move with the particle velocity of its spatial neighborhood, \underline{v} , plus its intrinsic diffusional velocity, \underline{v} , relative to the particle velocity. Thus the argument variables of the constituent functions f_S, f_{SL}, f_L, f_{LS} , and f_{LSL} are spatial points \underline{x} , at time, t , and species

FUEL AND CLADDING OXIDATION UNDER EXPECTED REPOSITORY CONDITIONS

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INTRODUCTION

Although its containment is imperfect, fuel cladding provides a redundant barrier that can help control release of radionuclides. However, if the disposal container fails while the fuel assemblies are still hot, both cladding and fuel could oxidize. Such oxidation could have a significant effect on long-term safety and doses to the public because oxidation increases the surface area of the fuel, exposure of the fuel to groundwater, and dissolution rate. This paper uses fuel temperatures and oxidation rates to determine the period during which oxidation is a concern.

WORK DESCRIPTION

The condition of a fuel rod will determine how the rod degrades. For intact rods, only the cladding is subject to oxidation. However, perforated rods can degrade by oxidation of the fuel as well. On exposure to oxygen at high temperature, fuel can oxidize to U_3O_8 . Such oxidation produces a large volume increase and would cause the cladding to split. Einziger¹ distinguishes two steps in this process: first, the fuel oxidizes in the vicinity of the perforation and causes a macroscopic split in the cladding; second, continued oxidation of the fuel near the ends of the split causes the split to propagate axially. Split propagation is neglected here because it has little effect on the total time for failure.

Einziger¹ has presented equations for these oxidation mechanisms. By combining the equations with information on fuel rod geometry for a Framatome Cogema Fuels Mark B4 assembly, equations giving characteristic times for degradation were derived.² For varying temperatures, the amount of damage is calculated by the method of damage accumulation.

Fuel temperatures will depend on the mass loading and the choice of backfill. For emplacement without backfill, temperatures were taken from calculations by Bahney³ for a typical pressurized water reactor fuel. These temperatures are plotted in Figure 1. Fuel temperatures for a backfilled drift were estimated by calculating the temperature drop for radial heat conduction from the waste package through a gravel backfill. The backfill acts as a thermal insulator.

RESULTS

If the disposal container fails while the fuel assemblies are hot, the rods will begin to oxidize. Figure 2 shows the fractional progress toward failure (cladding oxidation or split initiation) from the time of container failure until 1000 years. For example, for a backfilled drift and a mass loading of 20.5 kg U/m^2 , if the container fails at 100 years, 5% of the cladding will oxidize over the next 900 years, but if the container remains intact for 200 years, less than 1% will oxidize in the next 800 years. Oxidation at later times will be slow because of the low temperatures. The results show that cladding oxidation can be dismissed as a failure mechanism if the drifts are not backfilled; the temperatures are too low for significant oxidation. However, if the drift is backfilled at the time of emplacement and the fuel is immediately exposed to air, the cladding will oxidize completely.

If the cladding is perforated, longer protection is required to prevent split initiation. The time for which protection is required varies from about 35 years for a mass loading of 5.93 kg U/m^2 and no backfill (not shown in figure) to about 500 years for 20.5 kg U/m^2 with backfill. These results may appear to favor low mass loadings, but in fact high mass loadings are also acceptable. The temperature results show that while the fuel is hot enough for oxidation to be significant, the surface of the

LONG-TERM CORROSION BEHAVIOR OF ENVIRONMENTAL ASSESSMENT GLASS

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ABSTRACT

We have conducted static dissolution tests to characterize the corrosion behavior of the Environmental Assessment glass under conditions relevant to long disposal times. The glass dissolution rate decreased initially, but increased coincidentally with the formation of alteration phases including analcime, gmelinite, and a sodium aluminum silicate hydrate. However, the dissolution rate after those phases formed was less than the rate measured in a 7-day test.

I. INTRODUCTION

We have conducted static dissolution tests for durations up to about 18 months at high glass surface area/solution volume (S/V) ratios to characterize the behavior of Environmental Assessment (EA) glass¹ and to determine if the relative durabilities of a waste glass and EA glass measured in a short-term test provide insight into their relative long-term durabilities. Test conditions were the same as those under which alteration phases formed and increased the glass corrosion rate of other Defense Waste Processing Facility (DWPF) reference glasses.²

II. EXPERIMENTAL

Static dissolution tests were conducted at 90°C following the Product Consistency Test (PCT) Method B procedure³ from 56 to 527 days at S/V ratios of about 2000 and 20,000 m⁻¹ in a tuff ground water solution.²

Leachates were analyzed for pH, anions, and cations, and corroded glass and alteration phases were analyzed with optical microscopy, scanning electron microscopy (SEM) with energy dispersive X-ray spectroscopy (EDS), and X-ray diffraction (XRD).

III. RESULTS

The normalized mass losses, NL(i), are plotted against the test duration in Figs. 1 and 2 for tests conducted at 2000 and 20,000 m⁻¹, respectively. Boron has the highest normalized release and is used to calculate the glass corrosion rate. Differences in the calculated values of NL(i) indicate that Si, Li, and Na are partially incorporated into alteration phases. The corrosion rate of glasses in tests conducted at 2000 m⁻¹ increase from about 0.01 g/(m²·d) between 30 and 70 days to at least 0.1 g/(m²·d) after the precipitation of a zeolite alteration phase (see below).

Alteration phases had formed in all tests conducted at 20,000 m⁻¹. The rate measured over the first 200 days is about 0.1 g/(m²·d). The corrosion rate decreases at longer reaction times due to the loss of surface area as the glass corrodes. For PCT conducted with -100 +200 mesh crushed glass, the loss of surface area becomes significant when the value of NL(B) exceeds about 20 g/m². The nearly constant values of NL(B) in tests at 20,000 m⁻¹ indicate that glass is almost completely corroded after about 1 year.

EFFECTS OF TRANSPORT MODEL ALTERNATIVES INCORPORATING PRECIPITATION ON THE PERFORMANCE OF ENGINEERED BARRIERS

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ABSTRACT

The migration of radionuclide through bentonite was analyzed by alternative models considering the precipitation caused by decay-chain ingrowth. In the realistic model, the temporal and spacial isotopic ratio in bentonite was taken into account for determining the shared solubility for each radionuclide. The release rate of radionuclide from the outer surface of bentonite to surrounding rock is generally lower in such realistic analysis considering precipitation in bentonite than calculated by the model neglecting precipitation. This result shows the model not considering such effects is mostly conservative for the safety assessment.

I. INTRODUCTION

The precipitation caused by chain decay and ingrowth during the radionuclide transport through bentonite could reduce the effective thickness of bentonite for diffusion barrier, which in turn would increase the release rate of radionuclide from the engineered barrier system (EBS) of high level radioactive waste (HLW) disposal, therefore it is important to investigate the effects of precipitation.

The formation of precipitation fronts can be affected by several factors such as solubility limit and sorption coefficient. The selection of member radionuclides in the chains for the analysis could also affect the behaviour of precipitation.

The effects of precipitation fronts have been considered by relatively realistic models with different level of detail^{1,2,3,4}. This study discusses how those alternative conceptual models and parameter distribution would affect the radionuclide release from EBS in the Japanese HLW disposal concept.

II. MODEL ALTERNATIVES

This study investigates the perturbation for EBS performance caused by the alternative models and parameter variations focusing on the effects of precipitation fronts.

The following four models are considered in this study.

- Model 1 ; the elemental solubility limit with precipitations.(the reference model).
- Model 2 ; the shared solubility limit changing with time at the waste glass-bentonite interface only with precipitation.
- Model 3 ; the shared solubility limit changing with time and space by considering the four decay chains including short-lived nuclides;
Cm-248->Pu-244->Pu-240->U-236,
Cm-245->Pu-241->Am-241->Np-237->U233,
Cm-246->Pu-242->U-238->U-234,
Cm-247->Am-243->Np-239->Pu-239->U-235.
- Model 4 ; the elemental solubility limit without precipitation.

The models used in this analysis are based on one-dimensional, diffusive transport with linear, reversible and instantaneous sorption. The solubility limit and zero concentration boundary conditions are imposed at the waste glass-bentonite interface and at the bentonite - surrounding rock interface, respectively.

III. RESULTS AND DISCUSSION

Pu-239 and Np-237 are the totally dominant isotopes on their elemental concentration, even though the shared solubility is considered. This cause no significant differences of time-dependent release rates of Pu-239 and Np-237 from bentonite among models 1-3 as illustrated in Figure 1.

IMPORTANT PARAMETERS IN THE PERFORMANCE OF A POTENTIAL REPOSITORY AT YUCCA MOUNTAIN
(TSPA-1995)

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ABSTRACT

A total system performance assessment (TSPA) was conducted to determine how a potential repository at Yucca Mountain would behave. Using the results of this TSPA, regression was done to determine which parameters had the most important effect on the repository performance. These results were consistent with the current conceptual understanding of the repository.

I. INTRODUCTION

A total system performance assessment (TSPA) was conducted to evaluate the effect of alternative repository designs and alternative scenarios of natural-system behavior at the potential Yucca Mountain high-level waste repository¹. The repository's predicted behavior is quantified using a number of different performance measures, including the peak engineered barrier system (EBS) release rate over the first 10,000 years, and the peak dose at the accessible environment over the first 10,000 years and the first 1,000,000 years. Each of these performance measures is affected by many repository characteristics and parameters. It is important to devote the available resources toward understanding those parameters that have the most impact on the repository performance.

II. WORK DESCRIPTION

Latin Hypercube Sampling was used to sample the approximately 260 stochastic parameters in the TSPA-1995 process models, for 100 realizations of the parameter distributions. Based on simulations and expert judgement, about 25 of these parameters were selected as potentially being the most important. Stepwise regression was then used to select the most important of these parameters, based on how

much of the variability in the performance measures could be explained by subsets of these parameters.

Since the structure of the relationship between the model parameters and the performance measures is unknown, stepwise regression was performed on three transformations of the data: rank regression, log-linear regression and log-log regression. The log-log regression has the most intuitive appeal, since many of the relationships are multiplicative, so it is the only one presented here.

III. RESULTS

A. 10,000-Year Engineered-Barrier-System (EBS) Peak Release Rate

For the cases analyzed, ⁹⁹Tc solubility, infiltration rate, and spent-fuel dissolution rate are the three most important parameters. Spent-fuel dissolution rate is important because it determines the amount of radionuclides that are mobilized allowing them to be transported out of the EBS. ⁹⁹Tc solubility, in particular, is important because the ⁹⁹Tc dominates the EBS releases rates. Also, since the EBS rate is controlled by advection of the aqueous phase (water dripping from fractures onto the waste packages), infiltration rate is another important parameter.

B. 10,000-Year Peak Dose to the Accessible Environment

For the scenario analyzed, the top two variables are the matrix velocity in the Calico Hills, nonwelded, vitric, hydrogeologic unit ($v_{mat} - CHnv$) and the unsaturated-zone Darcy flux or infiltration flux (q_{inf}). The latter is important because the peak concentrations of the radionuclides do not

TESTING OF ABSTRACTIONS FOR TOTAL SYSTEM PERFORMANCE ASSESSMENT

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ABSTRACT

Total system performance assessment requires the explicit quantification of all the relevant processes and process interactions. However, process level descriptions of all the processes required for the evaluation of the total system performance is computationally impractical, thus requiring the abstraction of these process level models. In this paper the unsaturated flow abstraction methodology developed for the current iteration of the total system performance assessment for the potential repository at Yucca Mountain is presented along with a simple test problem.

I. INTRODUCTION

Characterization of flow and transport through unsaturated and saturated porous media requires the detailed modeling of underlying physical and chemical processes. However, the process-level description of the total system is inherently complex due to the coupling between various processes/sub-systems, parameter and model uncertainties, spatial and temporal variabilities, and multiplicity of designs and scenarios. This complexity makes a direct, process-level analysis for evaluation of the total system performance computationally impractical. Abstraction of these process-level models are therefore necessary to facilitate repetitive simulations required of the probabilistic analyses. The word abstraction is used to connote the development of a simplified/idealized process model (or a functional relationship) which reproduces/bounds the results of an underlying detailed process model. The objective of this paper is to present the unsaturated flow abstraction methodology developed for the current iteration of total system performance assessment for the potential repository at Yucca Mountain, and to demonstrate that the results of the detailed process model and the abstracted model are reasonably similar for a simple test problem.

II. METHODOLOGY

In this study, TOUGH2 is used for detailed process-level modeling and the Repository Integration Program (RIP) is used for abstraction modeling.^{1,2} The results of a two-dimensional flow model are compared against the 1-D simplification used in RIP. The travel times of an unretarded species from the repository base obtained using both TOUGH2 and RIP are used for comparison.

A 2-D cross section extracted from the LBL-USGS site-scale model was used for detailed process-level modeling of the unsaturated zone.³ Steady-state simulations were carried out for three different infiltration rates ($q_{inf}=0.1, 0.5$ and 1.0 mm/yr). Column 153 of the 2-D cross section (which is located within the proposed repository block) was chosen as a representative one-dimensional pathway. From the simulations, the ranges of the matrix pore velocity (v_m), and fractional fracture flux (F_f) for each of the hydrostratigraphic units below the proposed repository horizon was obtained. Using the velocity values, the advective travel time from the proposed repository horizon to the water table was calculated.

A 1-D pathway, consistent with that segment of Column 153 between the proposed repository horizon and the water table was setup within RIP. This pathway was divided into four layers as follows: TSw - 111.2 m, TSv - 8.4 m, CHnv - 80.7 m, CHnz - 121.2 m. The advective transport of the tracer pulse was simulated for all the three infiltration rates, using the expected values of v_m and F_f obtained from TOUGH2 for each hydrostratigraphic unit. The choice of a finite pulse (with $\Delta t = 100$ yr) was dictated by the need to start with a finite time step size which could then be progressively increased so as to efficiently simulate transport over extremely long time periods ($\sim 10^5$ years). The minimum and maximum arrival times of the pulse were determined by examining its breakthrough curve at the water table.

TOTAL SYSTEM PERFORMANCE PREDICTIONS (TSPA-1995) FOR THE POTENTIAL HIGH-LEVEL WASTE REPOSITORY AT YUCCA MOUNTAIN

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ABSTRACT

The management and operating contractor for the potential high-level nuclear waste repository at Yucca Mountain, Nevada, has recently completed a new performance assessment of the ability of the repository to isolate and contain nuclear waste for long time periods (up to 1,000,000 years). Sensitivity analyses determine the most important physical parameters and processes, using the most current information and models.

I. INTRODUCTION

The evaluation of the ability of the overall repository system to meet the performance and safety objectives specified in the applicable regulatory standards is termed total system performance assessment (TSPA). Besides providing a quantitative basis for evaluating the suitability of the site, such assessments are useful in defining the most significant processes and the information gaps and uncertainties regarding these processes and their corresponding parameters. TSPAs explicitly acknowledge the uncertainty in the process models and parameters and predict the impact of this uncertainty on the overall performance. The aim of any TSPA is to be as complete and reasonably conservative as possible.

TSPAs evolve with time. As additional site and design information is generated, assessments can be revised to become more representative of expected conditions and less conservative. Previous TSPAs of the Yucca Mountain site have been conducted in 1991 and 1993 (Barnard et al., 1992; Eslinger et al., 1993; Wilson et al., 1994; Andrews et al., 1994). The current analyses (M&O, 1995) build on these previous ones, using more recent information.

II. WORK DESCRIPTION

The specific goals of TSPA-1995 were to develop and apply more representative models of (1) the drift-scale thermohydrologic environment, including more reasonable estimates of relative humidity and temperature adjacent to the waste packages; (2) waste-package degradation, including improved corrosion models of the mild-steel corrosion-allowance material and the effects of cathodic protection on the corrosion-resistant material; (3) the engineered barrier system (EBS) design, including possible alternative backfills and barriers; (4) near-field unsaturated-zone aqueous flux (i.e., dripping water in the emplacement drifts); and (5) unsaturated-zone flow and transport, including the potential effects of fracture-matrix interaction.

III. RESULTS

The effect of model and parameter uncertainty on predicted results is evaluated by a number of sensitivity analyses. Alternative repository designs and alternative scenarios for natural-system behavior are considered, including (1) low and high water-infiltration rates through the unsaturated zone (0.01–0.05 mm/yr and 0.5–2.0 mm/yr); (2) low and high thermal load (25 MTU/acre and 83 MTU/acre); (3) alternative thermohydrologic models for the near-field environment; (4) three assumptions for the initiation of waste-package corrosion (relative-humidity controlled, temperature and relative-humidity controlled, and temperature and relative-humidity controlled with cathodic protection); (5) five conceptual models of EBS transport and water movement (drips directly on the waste form, drips on the waste container but not on the waste form, no drips or capillary-barrier effect, no drips and aqueous EBS transport of ^{129}I and ^{36}Cl , and no drips and aqueous EBS transport of ^{129}I and ^{36}Cl plus ^{14}C transport directly to atmosphere—the "diffusion-only" model); (6) fracture/matrix interaction in the

THE NATIONAL ACADEMY OF SCIENCES REPORT AND
ENVIRONMENTAL RADIATION STANDARDS FOR YUCCA MOUNTAIN

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ABSTRACT

The Environmental Protection Agency (EPA) has the responsibility of setting environmental standards for the potential repository at Yucca Mountain, Nevada. The Agency is formulating those standards. The background, status, and next steps are reviewed.

I. BACKGROUND

In 1985, EPA issued 40 CFR Part 191, *Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste*, as the generic standards for the subject wastes. In 1992, two Federal laws were enacted -- the *Waste Isolation Pilot Plant Land Withdrawal Act (WIPPLWA)* and the *Energy Policy Act (EnPA)*. The WIPPLWA, exempted Yucca Mountain from coverage under Part 191. The EnPA directed EPA to set standards specifically for the Yucca Mountain site. It also directed EPA to contract with the National Academy of Sciences (NAS) to conduct a study to provide findings and recommendations on the bases of the standards. The EnPA also mandated responses to three inquiries: (1) would a health-based standard for individuals protect the general public; (2) could active institutional controls essentially prevent breaching of the repository or increasing exposures; and, (3) can scientifically supportable predictions of human intrusion over 10,000 years be made?

II. STATUS

The NAS began work in 1993 and released its report, *Technical Bases for Yucca Mountain Standards*, in August 1995.¹ To get input from the stakeholders, including the Department of Energy, the nuclear industry, the State of Nevada, local and tribal governments, environmental groups, and members of the public, EPA opened a comment period on the report and held public meetings in Amargosa Valley (Nye County) and Las Vegas, Nevada, and Washington, D.C. The Agency has also met with several interested parties.

It is critical that EPA works particularly closely with another stakeholder, the Nuclear Regulatory Commission (NRC) to ensure that the standards can be implemented and to take advantage of the NRC information and data on Yucca Mountain. Also, it will help NRC understand EPA's intentions as they proceed to implementation. To assist in attaining these goals, EPA and NRC have each appointed staff-level liaisons.

III. THE NAS REPORT

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

- (a) use of an individual-risk limit;
- (b) that compliance be determined at the time of peak risk, within the period of stability of the geologic environment (about one million years);
- (c) against a risk-based calculation of the effects of human intrusion;
- (d) that the consequences of an intrusion be calculated;

REGULATORY PERSPECTIVE ON NAS RECOMMENDATIONS
FOR YUCCA MOUNTAIN STANDARDS

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ABSTRACT

This paper provides a regulatory perspective from the viewpoint of the potential licensee, the U.S. Department of Energy (DOE), on the National Academy of Sciences (NAS) report¹ on Yucca Mountain standards published in August 1995. The DOE agrees with some aspects of the NAS report; however, the DOE has serious concerns with the ability to implement some of the recommendations in a reasonable manner.

I. INTRODUCTION

In many areas the NAS recommendations are consistent with DOE thinking, as documented in the recommendations made by the DOE to the NAS² and to the U.S. Environmental Protection Agency (EPA).³ Those areas include the use of a health-based standard, the focus of a standard on the people in the vicinity of Yucca Mountain, the application of negligible individual risk, the appropriate basis of a quantitative evaluation of compliance being the mean of calculated results, and the caution against the application of subsystem performance requirements. In other areas, the NAS recommendations raise concerns related to the difficulty in implementing the resulting requirements. Four such concerns are discussed below.

II. TIME FRAME FOR COMPLIANCE

The NAS recommended compliance with a risk-based standard at the time of greatest risk within the limits imposed by the long-term stability of the geologic environment. It stated that the fundamental geologic regime at Yucca Mountain could be expected to remain predictable for approximately 1,000,000 years. The DOE considers that demonstrating compliance at any site, including Yucca Mountain, by comparing with numerical limits that extend

for hundreds of thousands of years may not be feasible, given the current regulatory environment. The DOE concurs with the EPA⁴ and the NRC⁵ that a period up to 10,000 years is a reasonable time frame for requiring meaningful quantitative projections. If time frames longer than 10,000 years are addressed in a standard, then the associated requirements should be qualitative, for comparison purposes, and should be used to gain insights into overall system performance. This is ultimately a policy decision. The DOE further notes that this concern with imposing a standard for very long time frames is consistent with the 1990 NAS "Rethinking" report⁶ which recommends de-emphasizing quantitative model predictions.

III. RISK LEVEL

In advocating a risk-based standard, the NAS did not recommend a specific level of acceptable risk; however, it recommended that the acceptable level of risk be established through rulemaking. The NAS suggested that a risk level in the range of 10^{-5} to 10^{-6} fatal cancers per year be used as a reasonable starting point for the rulemaking. The proposed exposure scenario to humans assumes that future humans can and will access contaminated groundwater, but will not test and treat their water supply. This assumption makes the calculation of health effects tractable, but it is very conservative. While it is impossible to quantify the exact degree of this conservatism, it should be recognized and factored into setting the risk limits for the Yucca Mountain standard. The DOE recommends that, given this very large conservatism, a range of 10^{-4} to 10^{-5} fatal cancers per year should be used as a starting point for the EPA's rulemaking.

THE YUCCA MOUNTAIN PROBABILISTIC VOLCANIC HAZARD ANALYSIS PROJECT

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The Probabilistic Volcanic Hazard Analysis (PVHA) project, sponsored by the U.S. Department of Energy (DOE), was conducted to assess the probability of a future volcanic event disrupting the potential repository at Yucca Mountain. The PVHA project is one of the first major expert judgment studies that DOE has authorized for technical assessments related to the Yucca Mountain project. The judgments of members of a ten-person expert panel were elicited to ensure that a wide range of approaches were considered for the hazard analysis. The results of the individual elicitations were then combined to develop an integrated assessment of the volcanic hazard that reflects the diversity of alternative scientific interpretations. This assessment, which focused on the volcanic *hazard* at the site, expressed as the probability of disruption of the potential repository, will provide input to an assessment of volcanic *risk*, which expresses the probability of radionuclide release due to volcanic disruption.

The possible volcanic hazard to the proposed repository is magma feeding a dike or surface eruption, which could ascend directly through the repository or erupt/intrude

nearby, compromising the integrity of the waste isolation system. The basic elements, or variables, that need to be assessed to define the hazard are the spatial distribution and recurrence rates of future volcanic events in the region. A variety of models have been proposed since the early 1980s to evaluate these elements. The PVHA captures uncertainties in both the interpretations and models of future volcanism and the relevant recurrence and spatial distribution data.

The primary steps in the project were: (1) selecting the expert panel members; (2) identifying technical issues; (3) eliciting the expert's judgments; and (4) compiling the results. The results of the hazard analysis are discussed in Youngs and others (this volume).

The principal participants in the volcanic hazard assessment were the ten members of the expert panel, who were elicited regarding the volcanic hazard at the site; the members of the Methodology Development Team (MDT), who designed and facilitated the study; and the technical specialists, who provided specialized data and interpretations to the expert panel. Members of the MDT

APPROACH, METHODS AND RESULTS OF AN INDIVIDUAL ELICITATION FOR THE VOLCANISM EXPERT JUDGMENT PANEL

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Probabilistic volcanic hazard assessment (PVHA) of future magmatic disruption of the Yucca Mountain site was completed as a participating member of the volcanism expert judgment panel conducted by Geomatrix Consultants for the Department of Energy. The purpose of this summary is to describe the data assumptions, methods and results of the elicitation and to contrast this assessment with past volcanism studies conducted for the Yucca Mountain Project. In previous studies, we have attempted to establish a range of probabilistic estimates of magmatic disruption using multiple alternative models that are *permissive* with the limited record of basaltic volcanic events. No attempt was made to discriminate or weight alternative models with respect to plausibility of magmatic processes or suitability to the tectonic setting of the region. Thus probability distributions for magmatic disruption of the site represent "consensus" distributions of all possible alternative models. In contrast, individual elicitations by the members of expert judgment panel members are allowed to incorporate preferred judgments in the assumptions required for probabilistic assessments, and to weight alternative models used to establish recurrence rates and disruption probabilities. This elicitation represents the input of one panel member; all the elicitations are integrated through a process of aggregation conducted by the Geomatrix methodology team.

Late Miocene and younger basaltic volcanic activity has occurred non-uniformly in space and time in the Yucca Mountain region (YMR), probably from partial melting of hydrous lithospheric mantle that may or may not be associated with low rates of extension.¹ Two cycles of small volume basaltic volcanism postdate voluminous Miocene silicic volcanism associated with the Timber Mountain caldera complex;² the youngest cycle of basaltic activity (< 5 Ma) is emphasized in the PVHA.

Multiple observations concerning the tectonic and volcanic history of the YMR are especially relevant to PVHA. Tectonic activity has waned in the YMR since the Miocene. Concomitantly there has been a large decrease (> factor of 30) in the erupted volume of basaltic magma since the Pliocene accompanied by a possible slight increase in

the frequency of volcanic events. Post-Pliocene basaltic volcanic activity in the YMR is typical of the less active, interior parts of the Great Basin. There has been time-space migration of basaltic volcanism during the Pliocene and Quaternary characterized by a southwestward migration of eruptive sites. Past sites of volcanic activity can be used to define zones of more likely future volcanic activity but individual sites of past events are poor predictors of sites of future events. Ascent of basaltic magma may be facilitated by deep seated structural features and volcanic centers are more common in alluvial basins than range interiors probably because the former represent sites of continuing but low rates of extension.

The PVHA for the Yucca Mountain region requires: (1) identifying zones of preferential occurrence of future volcanic activity, (2) developing methods to estimate the recurrence rate(s) of events in the zones, (3) estimating the disruption ratio for the zones and (4) applying these data to the Yucca Mountain site. Identification of zones of volcanic activity is accomplished through assessment of the distribution of volcanic events using event-distribution models (weighting 40%) and structural-tectonic models (weighting 60%). Regional background models are also used but are not integrated into the elicitation; they do however, provide a basis for assessing probability bounds. Event-distribution models are established through systematic examination of patterns of the distribution of basalt centers for intervals of 1.15, 5.05 and 9.05 Ma. The preferred event-distribution models include the Plio-Quaternary (not including the basalt of Buckboard Mesa) and Quaternary models of the Crater Flat Volcanic Zone^{3,4} (weighting 80%), the distribution area of the Younger Postcaldera basalt⁴ (weighting 0.15), and the distribution area of the Postcaldera basalt⁴ (weighting 0.05). The structural-tectonic models include Plio-Quaternary and Quaternary pull-apart models⁵ (weighting 60%), Walker-Lane structural models (weighting 25%), and northeast-trending structural models (weighting 15%). The probability of magmatic disruption of the Yucca Mountain site must be > than background rates for the southern Great Basin and YMR (1.4 to 3.3×10^{-9} events yr^{-1} and \leq upper

AN ASSESSMENT OF FUTURE VOLCANIC HAZARD AT YUCCA MOUNTAIN

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ABSTRACT

Preliminary results and methods of a volcanic-hazards assessment for the proposed high-level nuclear-waste repository at Yucca Mountain are given. The most significant hazards are potential intersection of the repository by a basaltic dike, or structural disruption associated with dike intrusion. Two approaches are taken, which give similar results: homogeneous volcanic-source zones and spatial smoothing. The preliminary computed probabilities of intersection of the Yucca Mountain repository by a basaltic dike are in the range 10^{-7} to 10^{-8} per year.

I. INTRODUCTION

During the past 10 million years, basaltic eruptions have characterized the Yucca Mountain Region (YMR). Phenomena associated with basaltic volcanism, including shallow dike intrusion and structural disturbance of the shallow crust above ascending dikes, are therefore most significant in volcanic-hazard assessment. Basaltic volcanoes of the YMR occur in clusters, and volcano locations as well as the orientations of cogenetic dikes have been influenced by the state of stress in the upper crust, which is extending through a combination of normal faulting and basaltic-magma intrusion.

The principal region of interest is the "area of most recent volcanism" (AMRV),¹ but expanded to include other volcanic centers and aeromagnetic anomalies within about 40 km of the repository that are younger than about 10 million years. Lithostratigraphic and geochronologic data from volcanic materials in the region of interest are the principal tools for determining the location, age and physical characteristics of volcanism in the YMR. Other Great-Basin volcanic fields also give insight on volcanic processes and event magnitudes.

II. VOLCANIC-EVENT DEFINITION

An event is defined as a cogenetic set of intrusives and extrusives that are products of a single magma batch. An event occurs within the geologically brief time it takes to inject a sheet of basaltic magma into the shallow crust and to solidify - decades to hundreds of years. The spatial

dimensions of an event are best constrained by the dimensions of individual basaltic dikes in the shallow crust: typically less than 2 km long and less than 2 m wide, but dike swarms may exceed 15 km in aggregate length. The area of tensile disruption of rocks above an ascending dike in the YMR is assessed to be about 2 km long by 0.5 km wide, and cogenetic volcanic materials will be emplaced mainly within this zone. Event magnitude (the area of magmatic disruption) is therefore of the order one square kilometer. Future dike orientation is taken to be $N25E \pm 30$ degrees, which is perpendicular to the contemporary direction of least horizontal stress in the upper crust.

The number of events and their uncertainties are assessed for each volcanic center in the region of interest, including the potential for undetected events. Event counts are developed for each of three homogeneous source zones.

III. TEMPORAL MODEL OF VOLCANISM

A homogeneous Poisson model is used because the data satisfy this model, and it has the important attribute of simplicity. The different time frames used in establishing each of the 10-, 5-, and 1-million-year homogeneous source zones are weighted to capture uncertainty about the time period that represents temporally homogeneous (and representative) magmatic events, and what these events might imply for the future. The post-10-Ma period captures post-silicic volcanism in the region of interest, permits incorporation of the 10-11 Ma Solitario Canyon basaltic dike near the repository, provides a background zone for the other, more localized zones, and is assigned the lowest weight because the long time frame is believed to be least representative of future volcanism. The post-5 Ma period does not mark a change of volcanic pattern within the region of interest, but is selected to provide an intermediate time frame for calculating a Pliocene-and-younger volcanic rate; it is given an intermediate weight. The post-1 Ma period is considered to be most representative of future magmatic processes, post-1 Ma volcanoes occur in the Crater Flat volcanic field near the proposed repository, and this time period is therefore heavily weighted. Any time-dependent differences in volcanic rates

RESULTS OF THE PROBABILISTIC VOLCANIC HAZARD ANALYSIS PROJECT

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The Probabilistic Volcanic Hazard Analysis (PVHA) project, sponsored by the U.S. Department of Energy (DOE), has been conducted to assess the probability of a future volcanic event disrupting the potential repository at Yucca Mountain. The methodology for the PVHA project is summarized in Coppersmith and others (this volume). The judgments of ten earth scientists who were members of an expert panel were elicited to ensure that a wide range of approaches were considered. Each expert identified one or more approaches for assessing the hazard and they quantified their uncertainties in models and parameter values. Aggregated results are expressed as a probability distribution on the annual frequency of intersecting the proposed repository block. This paper presents some of the key results of the PVHA assessments. These results are preliminary; the final report for the study is planned to be submitted to DOE in April 1996.

HAZARD MODEL COMPONENTS

The possible volcanic hazard to the potential repository is magma feeding a dike or surface eruption, which could ascend directly through the repository or erupt/intrude nearby, compromising the integrity of the waste isolation system. A volcanic *event* is defined in terms of temporal and spatial aspects. For the event definition used by several experts, an event occurs within tens to hundreds of years and is controlled by the process of magma ascent and crystallization. The expected spatial dimensions of an event are typically on the order of a one-meter wide basaltic dike of 1 to 5 km in length and, with lower probability, dikes in the range of 10 to 20 km in length. The future volcanic hazard is assessed by defining the spatial distribution and the recurrence rates of future volcanic events in the region.

Spatial Models:

Spatial models represent the future locations of volcanic activity. The most common models used are those that consider the future occurrence of volcanoes to be homogeneous within particular defined regions or 'source zones.' These source zones are defined based on several criteria: the spatial distribution of observed basaltic volcanic centers (especially post-5 Ma centers), structurally-controlled provinces, regions defined based on geochemical affinities, tectonic provinces, and other criteria. Parametric spatial distributions of future volcano occurrences were also given, such as the assessment that future volcanic centers will follow a bivariate Gaussian distribution within the Crater Flat volcanic field. Finally, spatial smoothing models were used by some experts to assess the relative likelihoods of future volcanic centers. These models essentially 'smooth' the locations of existing centers to obtain the likelihood function for future occurrences.

Temporal Models:

Temporal models define the frequency of occurrence of volcanic activity and, hence, the probability of occurrence. Many of the experts used homogeneous Poisson models to define the temporal occurrence of volcanic events, using the number of past volcanic events that have occurred during various periods in the past. Based on their considerable knowledge of the regional tectonic and geologic data, pertinent periods of the geologic record are identified, and individual volcanic events are interpreted and their timing is estimated. Non-homogeneous models were used by some experts to consider the possibility that volcanic events are clustered in time and/or to describe the possible waning or waxing

CRITICAL GROUPS FOR GEOLOGICAL DISPOSAL PERFORMANCE ASSESSMENTS

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INTRODUCTION

International recommendations and many national regulations on deep geological disposal of radioactive waste include requirements to demonstrate that radiation doses and/or risks to individuals after waste disposal meet particular quantitative standards. Therefore, an important question for performance assessments (PAs) is how to define the characteristics of the relevant individuals so as to allow their radiation exposure to be assessed. In the USA and in the context of high-level waste (HLW) disposal, this question has been given an increased profile following the publication of the National Academy of Sciences (NAS) report on technical bases for Yucca Mountain standards [1].

This paper presents a summary compilation and analysis of information on critical groups being prepared for the Electric Power Research Institute in order to inform judgements on assumptions for critical groups which might be used in PA of Yucca Mountain [2]. Consideration has been given to the regulatory development of the concept of a critical group and its use in radiological assessments, including assessments of present day effluent releases as well as releases which might occur in the long term future as a result of land based disposal of solid radioactive waste.

EXISTING NATIONAL AND INTERNATIONAL DEFINITIONS

The International Commission on Radiological (ICRP) used the term 'critical group' in ICRP Publication 26 [3] in 1977, which is probably far back enough in time to be a suitable starting point for discussion. ICRP's interest was in demonstrating compliance with dose limits. ICRP made it clear that they were not intending dose

limits to apply to the most highly exposed individual. Their intention was that the limits should apply to average exposure of members of a relatively homogeneous group whose exposure, because of their particular behaviour etc, should be representative of those individuals in the population expected to receive the highest dose. Because of the innate variability within an apparently homogeneous group, some members of the critical group could receive a dose higher than the mean. In 1985, ICRP [4] suggested that when a critical group cannot be readily identified, as in assessments of the long term future, a hypothetical group or individual should be considered who, due to location and time, would receive the greatest dose. This guidance remains essentially unchanged in more recent ICRP advice [5].

ICRP are not prescriptive about the habits or characteristics of the critical group. Rather they should be based upon present knowledge, and 'cautious, but reasonable', assumptions should be used. On the same issue, the Nuclear Energy Agency suggest [6] 'pessimistic but not unrealistic' assumptions.

Review of various national regulations and guidance shows that they all refer to ICRP and claim consistency with the ICRP radiation protection objectives [2]. Most guidance suggests that seeking to identify the most exposed person and their exposure is not a practical objective. It is not possible to determine actual behaviours, and to attempt to identify a hypothetical worst case leads to a slippery slope of increasingly pessimistic assumptions. In the extreme, the assumptions for behaviour become inconsistent with the metabolic basis for the definition of effective dose [5].

BIOSPHERE MODELLING FOR RADIOACTIVE WASTE DISPOSAL

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I ASSESSMENT CONTEXT

Swiss radiological protection regulations¹ specify dose and risk limits for radioactive waste disposal. Biosphere modelling is used to estimate doses to inhabitants of the potentially affected region. No time limit is specified and for biosphere modelling in the far future a reference biospheres approach is recommended. Predictions of future exposures are not made, but representative scenarios based on present day analogues are used to show that the potential radionuclide releases would not breach regulatory limits under these reference conditions. Probabilistic modelling may form part of the assessment but is not a formal requirement.

In scope and structure performance assessments require models which are simultaneously comprehensive and straightforward. FEPs (Features, Events and Processes) are limited by screening arguments to only those relevant for the particular assessment. Many biosphere FEPs can be ruled out by siting considerations but the parallel nature of pathways for environmental exposure (many FEPs with similar consequences, most important pathway changing in time) means that it is not possible, *a priori*, to define a general purpose biosphere model which is also very simple.

II. TAME: THE TERRESTRIAL-AQUATIC MODEL OF THE ENVIRONMENT

Experience in Switzerland and in international studies has shown that a good physical basis for the parameterisation is necessary for the construction of flexible and consistent models. The Swiss biosphere assessment model TAME² has been used in assessments of both L/ILW and

HLW in Switzerland^{2,3} and has been tested in international studies⁴.

A. Radionuclide Transport

A compartment model comprising radionuclide reservoirs representing an aquifer, deep-soil, top-soil, river water and river bed sediment is defined. Fractional transfer rates (*transfer coefficients*) define the interactions between compartments *i* and *j*:

$$\lambda_{ij} = \frac{1}{\theta_i + (1 - \varepsilon_i)\rho_i k_i} \left(\frac{F_{ij} + k_i M_{ij}}{l_i A_i} \right) y^{-1}$$

The transfer coefficients are defined in terms of the (site and nuclide specific) characteristics of the compartments:

A_i	m^2	area,
l_i	m	thickness,
ε_i	-	material porosity,
θ_i	-	volumetric moisture content,
ρ_i	$kg\ m^{-3}$	solid material density
k_i	$m^3\ kg^{-1}$	solid - liquid distribution coefficient for the contaminant (k_d).

Transport FEPs act either on radionuclides in solution (mediated by the water flux F_{ij} , $m^3\ y^{-1}$) or on those sorbed onto solid material (solid flux M_{ij} , $kg\ y^{-1}$). Site specific mass balance schemes are required for all water and solid fluxes. Water fluxes include rainfall, evapotranspiration, irrigation, infiltration, etc. Solid material fluxes include bioturbation, erosion, suspended solid load, etc. Thus, TAME is generally applicable to any situation where water and solid fluxes can be defined.

BIOSPHERE MODEL FOR ASSESSING DOSES FROM NUCLEAR WASTE DISPOSAL

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I. INTRODUCTION

In Canada's nuclear fuel waste disposal concept, the waste would be placed in corrosion-resistant metal containers, surrounded by clay-based buffer and backfill materials, in a vault deep in plutonic rock of the Canadian Shield¹. The engineered and natural barriers of the disposal system are designed to isolate the waste from the surface environment. Nevertheless, isolation may not be complete for all time and nuclides could reach the surface environment. Because this would likely occur far in the future, the impact on the environment and humans must be predicted with the help of mathematical models. The Atomic Energy Control Board (AECB), a key regulator of Canada's nuclear industry, requires that quantitative model simulations extend to at least 10,000 years². The AECB has established an individual risk limit for human exposure of 10^{-6} serious health effects per year. This limit corresponds to a radiological dose of 0.05 mSv/a or about 2.5% of the natural background dose, based on the AECB's risk conversion factor of 0.02. To demonstrate environmental and human safety, radiological doses are predicted to a member of a self-sufficient critical group, the most exposed people for up to 10,000 years³. For times longer than 10,000 years, reasoned arguments are required to show that no sudden or dramatic increases will occur that would be unacceptable by today's standards. Our predictions are based on linked vault, geosphere and biosphere models, which compose the system model.

The location for a nuclear fuel waste disposal vault on the Shield has not yet been determined; it must await approval of the disposal concept. Therefore, the biosphere model BIOTRAC, is very general and based on the Shield as a whole. BIOTRAC, is composed of integrated surface water, soil, atmosphere and food-chain and dose submodels^{4,5,6,7}. They quantify the behaviour and transport of radionuclides for dose

prediction. Modelling has been supported by an ongoing comprehensive research program to ensure the models are rooted in reality.

The objective of this paper is to summarize BIOTRAC and discuss how it is linked to the vault and geosphere models through a geosphere/biosphere interface. Other features presented are the critical group concept in the Canadian context and the inclusion of non-human dose receptors. Finally, presentation of a sensitivity analysis indicates the important pathways to man and other receptors for the major radionuclides released from the vault inventory.

A. Modelling Approach

1. Scenario Analysis and Systems Variability

First, important pathways and processes were identified and included in the submodels. The biosphere processes that control nuclide transport are fundamentally time dependent. In developing BIOTRAC, we have considered both fluctuating and transitional processes. For the former, the focus is mainly on distributed parameter values using probability density functions, PDFs, and for the latter, on model structure and PDFs. Through scenario analyses a combination of related factors (features, events and processes) were identified that could affect waste isolation. The most likely scenario is slow corrosion and failure of the waste containers, release of nuclides into the groundwater, migration through the buffer and backfill materials and through the geosphere to the biosphere.

The system model accounts for both uncertainties of predictions far in the future and the variability in the biosphere. The models are linked through SYVAC3, a SYstem Variability and Analysis Code⁸. At the beginning of each simulation run, SYVAC3 employs

BIOSPHERE FEP LIST DEVELOPMENT SPECIFIC TO YUCCA MOUNTAIN

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INTRODUCTION

Performance assessment (PA) methodologies for repositories for high-level waste (HLW) involve the collation and analysis of a wide variety of information obtained from many sources and involving many scientific and technical disciplines. The information is commonly described as Features, Events and Processes, or FEPs. Coherent application of all this information to the assessment has been a major component of methodology development. As a recent example, see the Swedish Nuclear Power Inspectorate report on systems analysis, scenario construction and consequence analysis definition for SITE 94 [1].

This paper summarises work being undertaken for the Electric Power Research Institute (EPRI) to develop a list of FEPs relevant to the biosphere component of a PA for HLW disposal at Yucca Mountain. The list is being developed taking account of a generic international FEP list produced within the BIOMOVs II study [2]. The objective of creating the list of FEPs is to provide a basis for conceptual models for radionuclide migration and accumulation in the biosphere following release from the geosphere and the evaluation of the corresponding radiation exposure of humans. A key issue is to provide a documented and justified explanation of how the information available for the assessment has been taken into account in assessment calculations. It includes justification for exclusion of some FEPs from calculations where this is appropriate. It also includes developing a basis for knowing when a line of investigation for a FEP has been taken far enough.

ASSESSMENT CONTEXT

In order to address the points raised in the introduction it is necessary to have an assessment context [2]. Such a

context will change as a disposal project proceeds. In general terms this will be because more site specific information will become available. For the biosphere part of a PA, it is especially important to be aware that regulatory requirements may change, eg concerning dose calculations. Equally important, the nature of the more likely and more significant releases of radionuclides from the geosphere will become clearer as a project evolves.

The main features of the assessment context are provisionally defined with example expansion as follows.

- **Assessment Purpose:** to demonstrate compliance with regulatory requirements.
- **Radiological endpoints:** dose to average member of a critical group(s). For further discussion see [3].
- **Geosphere release:** Radionuclide contamination in groundwater abstracted from a deep well. (It is recognised that other release mechanisms are possible.)
- **Human Society:** Technically developed, as now, consistent with ability to operate a deep well. (Again, it is recognised that other possibilities arise, eg for other societies to co-exist in the same location at the same time.)
- **Climate:** Substantially as present, since otherwise a well would not be required. (However, the need for consistency with assumptions for climate in other parts of the PA is recognised.)

ANALYSIS OF FEPS

There is insufficient space here to describe in detail how the FEPs are organised into a system to support conceptual model development, let alone list the FEPs