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Reports of
The Advisory Committee on Nuclear Waste

July 1999 - June 2000

U. S. Nuclear Regulatory Commission

September 2000

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ABSTRACT

This compilation contains 11 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the Twelfth year of its operation. The reports were submitted to the Chairman and Commissioners of the U. S. Nuclear Regulatory Commission (NRC). All reports prepared by the Committee have been made available to the public through the NRC Public Document Room, or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room); the U. S. Library of Congress, and the Committee's Web site at http://www.nrc.gov/ACRSACNW.

PREFACE

The enclosed reports are the recommendations and comments of the U. S. Nuclear Regulatory Commission's Advisory Committee on Nuclear Waste during the period between July 1, 1999 and June 30, 2000. NUREG-1423 is published annually. Volumes 1 through 9 contain the Committee's recommendations and comments from July 1, 1988 through June 30, 1999.

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NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

August 9, 1999

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Dicus:

SUBJECT: COMMENTS ON DOE'S LICENSE APPLICATION DESIGN SELECTION PROCESS (LADS) AND RECOMMENDED REPOSITORY DESIGN

This letter conveys our observations and recommendations regarding the Department of Energy's (DOE's) License Application Design Selection (LADS) process and the Management and Operations Contractor (M&Os) recommended repository design for the site recommendation (SR) and license application (LA). The letter also transmits the attached "white paper" by Charles Fairhurst titled, "Engineered Barriers at Yucca Mountain -- Some Impressions and Suggestions." In his white paper, Dr. Fairhurst examines some geotechnical aspects of the repository design in the setting of Yucca Mountain with particular attention to two issues - (i) reduction of water inflow to the waste emplacement drifts and (ii) pre- and post-closure stability of the drifts. A concept of an innovative repository design not presently being considered by the DOE is described, together with some impressions of the currently favored repository design. We hope that the paper will help the NRC as it prepares to conduct a thorough and critical safety review of the final repository design and the projected overall performance of the Yucca Mountain high level waste (HLW) disposal facility.

The observations and recommendations we make here are based on briefings we heard on July 20, 1999 on DOE's license application design selection, during the 111th ACNW meeting in Rockville, Maryland. The basis for the attached white paper is derived from a variety of sources, including the DOE's viability assessment, and interactions with the NRC and DOE staffs, the Center for Nuclear Waste Regulatory Analyses (CNWRA), the M&O, the ACNW, and others.

White Paper on Engineered Barriers at Yucca Mountain

In the attached paper, Dr. Fairhurst examines a repository shield concept that appears to have the potential to greatly reduce water infiltration into repository drifts. The shield acts like an umbrella above the repository to divert water around drifts by taking advantage of the vertical fractures and predominantly vertical flow system in the vicinity of the repository horizon. The shield system may also help reduce near-field flow uncertainties in designs such as the Enhanced Design Alternative-II (EDA-II) currently recommended by the M&O to the DOE. The shield concept is shown to be most effective when used in conjunction with a multi-layered repository to minimize the surface area contacted by infiltration. Dr. Fairhurst suggests that if the shield can be demonstrated to be effective with high confidence, it may be possible to avoid the need for the very costly ($4.6 billion) titanium drip shield used in the EDA-II.

The purpose of the paper is not to promote or endorse a specific design. Rather, the paper is intended to demonstrate that there may be innovative ways to engineer the natural setting such that the overall performance of the repository is improved. Current DOE designs appear to concentrate exclusively on engineering options within the drift itself. We believe that exploration of such ideas supports the NRC in its mission and in its vision of "enabling the safe and efficient use of nuclear materials." Consideration of the repository shield and a multiple level repository and other design concepts can provide insights into approaches for reducing critical uncertainties and for modifying the degree of reliance placed on natural versus engineered barriers. Exploration of alternative design concepts may also provide insights to help the NRC avoid placing constraints on DOE's repository design that might inadvertently limit possible future beneficial design changes and innovations, that would lead to greater confidence in the safe disposal of HLW at Yucca Mountain.

In its July 9, 1999, letter to Lake Barrett (DOE)[1], the Nuclear Waste Technical Review Board (NWTRB) expresses concern about the uncertainties associated with the above-boiling-temperature EDA-II design recommended by the M&O, and the lack of transparency in the process and rationale used to select this design. The EDA-II design is a "high temperature" design having a peak drift-wall temperature (160°C) above the local boiling point of water (96°C), with the space between drifts below boiling. To reduce uncertainties, the NWTRB urges DOE to consider modifying the EDA-II design to achieve below- boiling temperatures everywhere in the rock by increasing the rate or duration, or both, of ventilation before repository closure.

The ACNW believes that further analyses must be done before a determination can be made on a choice between a "totally below boiling" temperature repository and one in which some boiling takes place. Dr. Fairhurst points out that the recommended EDA-II design has some merits but also some disadvantages. Although a cooler repository design may simplify modeling of water redistribution, the potential for a higher temperature repository design to reduce the quantity of water reaching the drifts should not be abandoned without further assessment. It is possible that the existing EDA-II design, possibly modified to include multi-layered emplacement drifts, in conjunction with the infiltration shield concept, can be shown to reduce the uncertainties of water refluxing associated with a hot repository while maintaining the advantage of the hot repository to drive moisture away from the canisters.

We hope that you find Dr. Fairhurst's white paper to be of interest.

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[1]July 9, 1999 letter from Jared L. Cohen, Chairman, Nuclear Waste Technical Review Board, to Lake H. Barrett, Acting Director, Office of Civilian Radioactive Waste Management, U.S. Department of Energy.

Observations and Recommendations Regarding the DOE's Design Selection Process and the Recommended Repository Design

Observation 1

Over the past 10 months, the M&O contractor has been conducting a study of alternative repository designs for the proposed Yucca Mountain repository. As noted earlier, the M&O recently recommended that DOE select the EDA-II. The DOE has not yet made a decision about adopting the M&O's recommendation. The recommended EDA-II design differs significantly from the repository design presented in the DOE's viability assessment. As noted above, the NWTRB has expressed its dissatisfaction with the design selection process as well as with the recommended EDA-II design. Such recent and rapid changes suggest that the fundamental design and the many design-related details are likely to continue to change until such time as DOE submits its LA to the NRC. DOE's repository design must be regarded as a work in progress.

Recommendation 1A:

The NRC should plan for continued change in the repository design up until the time the LA is submitted. It follows that the NRC staff should adopt realistic expectations about the turnaround time that may be required to conduct a thorough review of the SR or LA design. The NRC should also develop a license review strategy that allows the DOE maximum flexibility to implement beneficial design changes and other innovations before its submittal of the LA as well as times throughout the preclosure period of the repository.

Recommendation 1B

As noted in the attached white paper, the preclosure period of the repository could last as long as 300 years, and, because of this, the NRC staff must be careful to avoid placing constraints on the design that might preclude future beneficial design changes or innovation. The NRC staff must ensure that it is prepared to recognize such innovation during its review of the LA. Further, as part of a strategy to develop review capability and insights into repository systems, the NRC and the CNWRA staffs should conduct independent evaluations of alternative, cost-effective designs. In evaluating such innovative designs as part of its preparation to review the LA, the NRC staff would gain insights into the relative importance of various design features, alternative strategies to reduce critical uncertainties, and alternative strategies for demonstrating defense in depth. The insights gained through the evaluation of alternative design concepts will enhance the NRC staff's capability to assess repository safety.

Observation 2

NRC's proposed rule governing HLW disposal (10 CFR Part 63) requires monitoring of repository performance. The 50- to 300-year repository preclosure period presents a major opportunity to establish the validity of design assumptions. Monitoring will require "performance confirmation drifts"[2]. Such drifts, appropriately located, could also serve as part of the flow diversion system proposed in the white paper.

Recommendation 2

The ACNW endorses the sentiment expressed recently by the U.S. Geological Survey (USGS), "that a careful description of the proposed monitoring strategy, as well as a detailed and complete list of what is to be monitored—and why, where, how, and for how long—should be developed expeditiously." [3] We encourage the NRC staff to consider long-term monitoring needs and strategies for how DOE may factor performance confirmation monitoring into its final design.

Observation 3

As noted above, in its July 9, 1999, letter to L. Barrett, the NWTRB expresses concern over the lack of transparency in the assumptions and value judgments made in the design selection process as well as the recommended design. Implicit in the NWTRB's letter is that the Board is uncomfortable with the M&O's selection of the EDA-II repository design because of the current uncertainties associated with high repository temperatures. It is not clear to the ACNW how the uncertainty associated with the various design concepts and features has been quantified and factored into the M&O's process for selecting a preferred design. The M&O's identified evaluation criteria do not include uncertainty as a criterion for making a selection. The conceptual model and assumptions for the various design concepts and features will drive the results of the evaluation and comparison of alternatives.

Recommendation 3

The ACNW believes that the M&O's approach used to evaluate and compare quantitatively the various EDAs has not been made transparent. We encourage the NRC to ensure that the rationale, approach, and assumptions used in the evaluations and in comparisons of alternatives are appropriate. In addition, as noted in recommendation 1B, the NRC and CNWRA staffs should conduct their own independent evaluations of alternative, cost-effective designs, similar to the evaluation of the innovative design described in the attached white paper.

Sincerely,

B. John Garrick
Chairman

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[2]Viability Assessment of a Repository at Yucca Mountain, Preliminary Design Concept for the Repository and Waste Package, USDOE, Volume 2, 1998, p. 4-111

[3]USGS Circular 1184, 1999, "Yucca Mountain as a Radioactive Waste Repository."

Summary

Yucca Mountain was initially recommended as a potentially suitable site for a high-level waste repository because it was anticipated that it would be dry. The repository would be situated in the unsaturated zone at a depth of 300 m below the surface and approximately 300 m above the current water table. It was also proposed as a "hot repository," in which rock temperatures would rise above 200 °C and would remain above the boiling point of water for several thousands of years. The intent was to prevent any liquid water from reaching the waste packages during that period.

Recent studies suggest that infiltration rates in the unsaturated zone may be higher than originally anticipated, and may increase substantially 20,000 years or so into the future. This information has prompted a redesign of the repository placing greater emphasis on engineered barriers within the waste emplacement drifts, e.g., a drip diversion (Richards) barrier; corrosion-resistant waste package; titanium drip shield (cost $4.6 billion); active ventilation during the 100- to 300-year preclosure period; and lower repository temperatures.

The viability assessment (VA) published by the U.S. Department of Energy (DOE) in December 1998 indicates that these engineering measures should suffice to meet the 10 CFR Part 63 requirements of the U.S. Nuclear Regulatory Commission (NRC) over the 10,000-year regulatory period, although doses are predicted to rise considerably beyond 10,000 years.

These notes, prepared after review of the VA, focus on geotechnical aspects of the repository design. The author has profited from discussions with colleagues of the Advisory Committee for Nuclear Waste (ACNW) and NRC, as well as from participation in numerous meetings and discussions with staff of DOE and its Management and Operating (M&O) contractors. The notes emphasize (1) a repository shield concept and (2) prediction of drift stability during both the (100 yr ~ 300 yr) preclosure and postclosure periods.

This paper does not promote or endorse any specific repository design. Rather, its purpose is to stimulate the NRC's thinking as it prepares to conduct a thorough and critical review of the repository design used in DOE's license application. The paper attempts to demonstrate that consideration of such innovative ideas as the repository shield concept and triple-layer repository can redefine the problem by reducing or eliminating critical uncertainties, or altering the degree of reliance placed on natural versus engineered barriers.

Given that decisions regarding final closure will not be made until the end of the operational period of the repository, the NRC must be careful to avoid placing constraints on the project now that would inadvertently limit possible future advantageous design changes and innovation. It is incumbent on the NRC to have the capability and be prepared to recognize the possibilities for such innovation during its evaluation of the license application.

The repository shield acts as an *umbrella* above the repository, taking advantage of the (dominantly vertical) fracture and flow system of the site to divert water away from the

repository drifts. The shield uses natural material (rock) only, augments an existing design, can be developed at any time during the preclosure period, and can serve to house a remote-monitoring network for the repository.

For a repository shield to be most cost effective, the repository should be a multi-level (three-tier or two-tier) design. (Figure 2 shows a three-tier design.) The shield appears to have the potential of greatly reducing water infiltration to the repository drifts —with attendant reduction of doses and simplification of performance assessment calculations. Construction of a flow diversion barrier in the (radiation-free) slot excavations above the drifts would be simpler than remote placement around the unshielded waste packages in the repository drifts —as currently proposed by DOE. If water infiltration is reduced to the extent predicted by analysis to date (see Appendix I), the expensive titanium drip shield (see Figure 4) may not be required. The presence of the drainage slots directly above the emplacement drifts may also simplify near-field fluid-flow and reflux processes during the thermal cycle. The concept deserves serious examination by DOE and its contractors.

With respect to drift stability, the repository environment is unique in that substantial thermo-mechanical stresses may be generated in both the reinforcement support and the rock. From information available on the mechanical properties of the Topopah Springs formations, it appears that stable excavations can be designed in both the lithophysal and the non-lithophysal units. It is believed that rock reinforcement using fully grouted bolts, mesh, and (if possible) shotcrete is preferable to the use of concrete or steel set supports for the repository drifts. Attention will need to be given to pH control of the cement used, but this problem does not appear to be an insuperable problem.

For the postclosure period, it must be assumed that any rock reinforcement or support system will no longer be effective. Recent developments in the numerical modeling of long-term progressive degradation of the mechanical properties of rock masses can provide more realistic assessment and prediction of the behavior of rock around excavations that are not back-filled than were possible in the past. Progressive disintegration and collapse of the rock may, in fact, result in a "natural back-filling" process that could be as effective, eventually, as standard back-fill. Of course, this does not preclude the use of a "chemically tailored" back-fill in the drift section below the waste packages, which could provide significant radionuclide "capture" benefits.

Introduction

The goal of geological isolation of highly radioactive waste is fundamentally simple — to place the waste at depth in the subsurface such that the radioactive elements or *radionuclides* in the waste will never return to the biosphere in concentrations sufficient to pose a significant health risk to humans.

Given the very long half-life of some radionuclides, the times for which isolation is required may be on the order of several hundreds of thousands of years.[1]

The primary vehicle for transport of the radionuclides from the initial underground location or repository is moving water that comes into contact with the waste. Radionuclides become entrained in the water (by dissolution or by colloidal suspension) and move to the biosphere, either directly or in water that is pumped from the aquifer and used for drinking and/or irrigation.

Thus, one of the main criteria in repository siting is to minimize the probability of radionuclide uptake by water and transport to the biosphere. Some radionuclides have very low solubility in the groundwater, others may be very soluble. The physical and chemical characteristics of the rock may also greatly retard the overall rate of movement of particular radionuclides in relation to the rate of groundwater movement. The concentration may also be reduced by dilution (e.g., in water or air) so that release to the biosphere via large bodies of water (i.e., seas or oceans) can also provide an added measure of safety.

The first formal report on the feasibility of geological disposal was published by the U.S. National Academy of Sciences/National Research Council in 1957 (NAS/NRC, 1957). The report noted that:

> *Wastes may be disposed of safely at many sites in the United States, but, conversely there are many large areas in which it is unlikely that disposal sites can be found, for example, the Atlantic Seaboard. The research to ascertain feasibility of disposal has for the most part not yet been done . . . .*

The report concludes with the following two *General Recommendations on Corollary Problems:*

> 1. *The movement of gross quantities of fluids through porous media is reasonably well understood by hydrologists and geologists, but whether this is accomplished by forward movement of the whole fluid mass at low velocity or whether the transfer is accomplished by rapid flow in "ribbons" is not known. In deep disposal of waste in porous media it will in many cases be*

[1] The "half-life" of plutonium 239, for example, is 24,000 years, i.e., the specific radioactivity will decline to $(\frac{1}{2})^{10}$ (i.e., 0.001 or 0.1%) of its initial activity in $24{,}000 \times 10 = 240{,}000$ years, and to (0.001)(0.001) or 0.0001% in 480,000 years. Other very long-lived radionuclides that contribute to the potential dose at various (long) times at Yucca Mountain are technetium 99 (half-life of 212,000 years), uranium 234 (245,000 years), neptunium 237 (2.14 million years), and iodine 129 (17 million years).

essential to know which of these conditions exists. This will be a difficult problem to solve.

2. *The education of a considerable number of geologists and hydrologists in the characteristics of radioactive wastes and its disposal problems is going to be necessary.*

Today, more than 40 years later, there are many hydrologists and colleagues in related disciplines worldwide who have studied groundwater flow in considerable detail. Significant advances have been made, but characterization of water flow still involves large uncertainties, especially in fractured rock masses. It remains "a difficult problem to solve."

Geological repository siting and evaluation programs are currently underway in approximately 30 countries. Of these, all but the Yucca Mountain project in the USA are sites below the groundwater table. For these, the host rock is usually of low intrinsic permeability with a low regional hydraulic gradient (i.e., the overall rate of water movement from the repository is expected to be very low). A number of countries are considering repositories in crystalline rock. Characterizing groundwater flow in fractures is frequently a serious issue for these sites.

In addition to understanding the natural system at Yucca Mountain, i.e., groundwater flow and radionuclide transport, NRC's proposed high level waste (HLW) disposal regulation, 10 CFR Part 63, indicates that *an engineered barrier system (EBS) consisting of one or more distinct barriers is required in addition to natural barriers.* The proposed rule states that *the Commission continues to beleive that multiple barriers, as required in the Nuclear Waste Policy Act of 1982 (NWPA), must each make a definite contribution to isolation of waste at Yucca Mountain.* Thus, DOE must design and demonstrate quantitatively that the total repository system relies upon and balances the contributions of both natural and engineered barriers to isolate waste.

The preclosure period of the proposed Yucca Mountain repository is expected to range from 50 to 300 years. Given that final repository closure will not occur until the end of the preclosure period, the NRC must be careful to avoid placing constraints on the project now that would inadvertently limit possible future beneficial design changes and innovations. It is incumbent on the NRC to have the capability (and be prepared) to recognize the possibilities for such innovation during its evaluation of the license application. One way to develop such capability is for the NRC to conduct an independent evaluation of viable, cost-effective designs. To conduct such evaluations, the NRC needs to have competent scientific and engineering expertise available over the broad spectrum of disciplines involved in repository design and long-term performance assessment. With the much larger complement of technical staff available to DOE and the recent and rapid changes in repository designs proposed by the DOE, the NRC faces a formidable challenge.

This report focuses on geotechnical aspects of the proposed Yucca Mountain repository. A design concept consisting of a repository shield used in conjunction with a multi-tiered repository is outlined. Particular attention is given to two issues: (1) diversion of groundwater before it reaches the waste-filled drifts and (2) drift stability. The paper then considers prediction of drift stability during the preclosure and postclosure repository periods. The paper compares the respository shield concept to the DOE's current, preferred repository design, which has changed significantly from the design presented in the DOE VA. The purpose of the paper is to stimulate the NRC's thinking as it prepares to conduct a thorough and critical review of the repository design used in DOE's license application. The ACNW may also use the ideas in the paper in preparing its specific comments on the DOE site recommendation and license application. The paper attempts to demonstrate that consideration of alternative, innovative design concepts, such as the repository shield/multiple-layer repository, may take better advantage of the geological characteristics of the proposed repository site at Yucca Mountain. Critical, persistent uncertainties may possibly be reduced substantially and the degree of reliance placed on natural and engineered barriers can be varied. The proposed "shield drifts" can also serve the role of performance-confirmation monitoring drifts (see VA, Vol. 2, p. 4-111).

Groundwater Flow at Yucca Mountain

At Yucca Mountain, the proposed repository horizon is in the unsaturated zone, approximately 250—300 m below the surface of the Amargosa Desert and 300 m above the water table. Tectonically, the region is currently undergoing extension (i.e., the rock mass is tending to extend horizontally). This implies that, at least near the surface (i.e., within the region of concern with respect to the repository), the lateral stresses in the rock are less (~3 MPa) than the vertical (gravitational or *overburden)* stresses (~7 MPa at a depth of 300 m). This situation has given rise to high-angle (i.e., almost vertical) fracturing (see VA, Vol. 2, Figure 2-9, p. 2-17). As a result of this situation, the fractures tend to be highly transmissive, so that rainfall and surface waters drain rapidly through the fractured mass into the groundwater. However, these fractures are generally not single, continuous planar features. Individual fractures are of limited extent, so that connected pathways, allowing flow through the fracture network, will be considerably less frequent than the individual fractures.

In initial planning for the repository (Roseboom, 1983), it was felt that the annual percolation flux (i.e., precipitation less the amount of surface evapo-transpiration) was very small (on the order of 1 mm/yr) and that little or no moisture would drain into the repository (i.e., the repository would be "dry"). In addition, it was decided to adopt a "hot repository" design (i.e., such a disposal layout that the rock temperature in the vicinity of the repository would remain well above 96 °C, the local boiling point of water, for hundreds or thousands of years, so that no liquid water could reach the waste canisters[2]).

More recent studies indicate that the total infiltration may be higher, and that a considerable portion of this may flow through the interconnected fracture pathways. As noted in the VA:

> *Estimates of average percolation flux from these various studies range from about 0.1—18 mm (0.004—0.7 in) per year. Because of Paintbrush attenuation most of the flux probably requires hundreds to thousands of years to reach the repository horizon. However, isotopic (chlorine-36) data suggest that at least a fraction of the flux reaches the repository level in ten years or less. Thus, while some of the*

[2] The high-temperature design is feasible in an unsaturated high permeability zone, such as exists at Yucca Mountain, where the pressurized water vapor in the rock in the vicinity of the excavations can "leakoff" readily toward the surface.

water moves downward quickly, much of it travels more slowly. (VA, Vol. 1, p. 2-38).

Studies of long-term climate change in the Yucca Mountain region over the past 500,000 years (see Figure 1) indicate that the climate in the region will very likely become colder *within the next few hundreds or thousands of years* (VA , Vol. 1, p. 2-30). Annual precipitation and infiltration are then likely to increase considerably. DOE performance assessment calculations consider a mix of dry and wetter climates extending up to several hundreds of thousands of years into the future (VA, Vol. 3, Sect 3.1.2.1, p. 3-15). These periods include dry climate conditions, as now, with an assumed base infiltration rate of 8 mm/yr, a long-term average period with a base infiltration rate of 42 mm/yr; and *superpluvial* periods with a base infiltration rate of 110 mm/yr (VA, Vol. 3, Table 3-5, p. 3-15). Increased infiltration rates will increase the proportion of total flow through fractures.

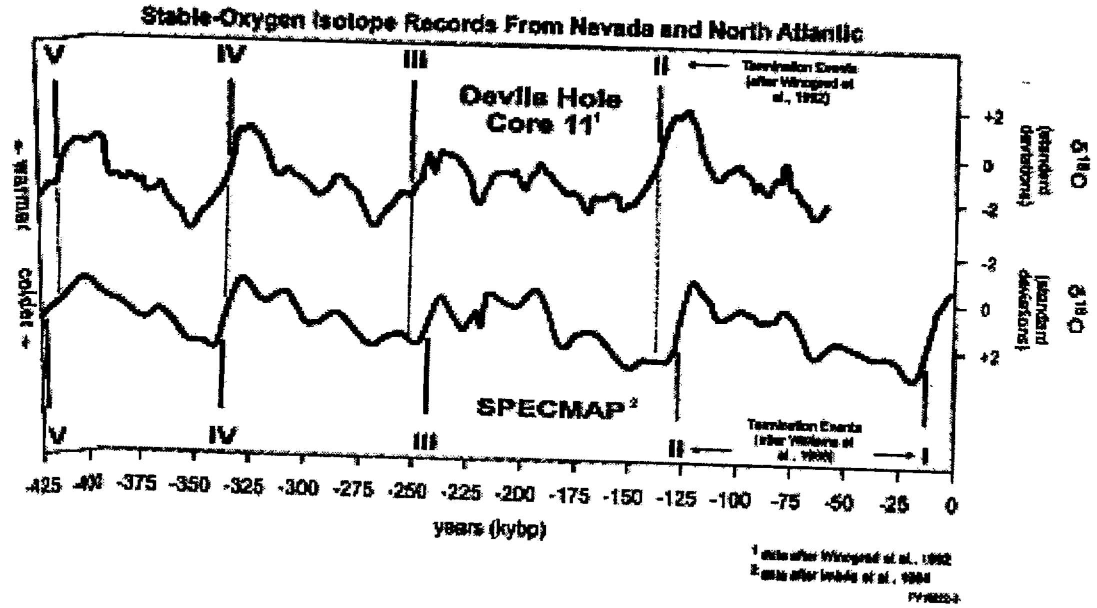

Figure 1 ***Stable-Oxygen Isotope Records from Nevada and North Atlantic as Indicators of Past Climate Variation in the Vicinity of Yucca Mountain***

The overall conclusion with respect to repository design at Yucca Mountain is that a significant fraction of the total infiltration through the unsaturated zone will be by flow through interconnected fracture pathways. The precise location of these pathways cannot be predicted, and the amount of flow may vary considerably from place to place in the repository. The rates of flow in these fracture pathways can be high, on the order of tens of meters per year.

A fraction of the flux arriving at the drift horizon is assumed to drip onto the waste packages, causing corrosion of the package and, eventually, contact with and dissolution of some of the waste. Details of the calculation procedure are outlined in the viability assessment (VA, Vol. 3, Sec. 4.1.3, p. 4-4 et seq.).

Repository Design and Yucca Mountain

Waste isolation poses unique problems for both geoengineering and geoscience. These problems center around the time frames involved, with at least semi-quantitative answers needed over times on the order of 10^4 or 10^6 years — far longer than the 10^1 or 10^2 years for which engineers are accustomed to provide quantitative solutions. The geoscience issues have received more attention to date, so there is a good awareness of the uncertainties associated with predictions presented with respect to waste isolation over such times. With engineering design now receiving more attention, it is important not to overlook the time element. Repository design considerations place severe constraints on the use of "engineering experience" and require an unprecedented reliance on predictive (often numerical) analysis.

Development of a convincing prediction of the performance of a waste package alloy thousands of years into the future, when that material may have been known for less than 100 years or so, is an example of the challenges involved.

Time Frames of Concern in Repository Design

The following three periods of interest can be distinguished in the design and assessment of long-term performance of a repository at Yucca Mountain:

> **Preclosure**[3] — Between 100 and 300 years (i.e., the period from the start of repository excavation until the decision is made to "close" the filled repository). Although it would not be impossible to retrieve waste from the closed repository, retrievability at Yucca Mountain is currently envisaged to be accomplished only during the preclosure period. The drift support system should be designed for the preclosure period.
>
> **10,000 years beyond closure** — This is the regulatory period specified in 10 CFR Part 63. If the total system performance assessment (TSPA) computations presented in the license application submitted by DOE are deemed by NRC to provide reasonable assurance that individual doses to a reference *critical group* located 20 km from the

[3] The 300-year upper limit was apparently chosen because it corresponds to ten half-lives of radioactive decay for cesium 137 and strontium 90.

repository do not exceed allowable limits at the end of 10,000 years after closure, the repository can be licensed.

Beyond 10,000 years — Although this period is strictly not part of 10 CFR Part 63, DOE acknowledges in the VA that doses will continue to increase significantly beyond 10,000 years, approaching the order of natural background radiation (Fig. 4.12 in VA, Vol. 3 shows a peak dose of 0.2 rem, at 200,000~300,000 years), almost an order of magnitude greater than the 25-mrem maxmimum dose allowed during the 10,000-year NRC regulatory period.

The U.S. National Academy of Sciences/National Research Council 1995 report, *Technical Bases for a Yucca Mountain Standard* (TYMS, 1995), recommended that the regulatory period be sufficient to cover the period of peak dose. As noted above, this period extends well beyond 100,000 years.

Some estimates indicate much higher doses than those given in the VA, as is illustrated in the following extract from a recent article by Carter and Pigford (1998)[4]:

Calculations by the project show that in 10,000 years the annual dose from drinking contaminated water from the repository will be about 0.02 rem per year. When the dose from eating food contaminated by irrigation water from these same wells is added, the total dose will be about 0.13 rem. This is 13 times the annual dose limit established by the U.S. Nuclear Regulatory Commission (NRC) two decades ago for persons living near a nuclear power plant. It is five times the

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[4] Pigford, T. H., and E. D. Zwahlen, " Maximum Individual Dose and Vicinity-Average Dose for a Geologic Repository, "Scientific Basis for Nuclear Waste Management XX, W. J. Gray and J. R. Triay, Eds, Materials Research Society, Pittsburgh, PA, 1996, Vol. 465, pp. 1099-1108 .

Professor Pigford also recently provided the writer with the following details concerning the doses mentioned in the quotation:

For the dose calculations, we relied first on the dose calculations in TSPA-95 (Akins, J. E., J. H. Lee, S. Lingineni, S. Mishra, J. A. McNeish, D. C. Sassani, S. D. Secoughian, "Total System Performance Assessment — 1995: An Evaluation of the Potential Yucca Mountain Repository," TRW, November 1995.) These doses were calculated only for drinking contaminated well water. Additional doses from food chains were not included in TSPA-95. We utilized the graphs showing the cumulative complementary distribution functions for 1,000,000 years and for 10,000 years. We selected the drinking-water doses at a CCDF of 0.05, corresponding to a 95% confidence level. The 95% confidence level is commonly used in engineering practice, it has been recommended by Britain's NRPB, it was recommended in my dissent appearing in the National Research Council's TYMS (1995) report, and it was incorporated in draft legislation proposed by Congress for Yucca Mountain.

From other graphs in TSPA-95 we identified which radionuclides were the principal contributors to these doses. From EPRI data (Smith, G.M., B. M. Watkins, R. H. Little, H. M. Jones, A. M. Mortimerk, "Biosphere Modeling and Dose Assessment for Yucca Mountain," EPRI Report TR-107190, 1996) we derived the ratio of total individual dose to drinking-water dose for each of the principal radionuclide contributors. Multiplying the drinking-water doses derived from TSPA-95 by the appropriate ratios yielded the doses reported in our article in the Bulletin of Atomic Scientists.

two decades ago for persons living near a nuclear power plant. It is five times the annual dose the NRC allows for persons making unrestricted use of a nuclear facility whose license has terminated. (The dose calculations allow a 5 percent probability of doses higher than those cited here.)

After 10,000 years, the calculated annual dose at a well three miles distant rises rapidly. Indeed, after 30,000 years, the annual dose from iodine 129 and technetium 99 will have increased about 80-fold, to 10 rems. Then the longer-term annual dose from neptunium 237 appears and rises to about 50 rem by about 100,000 years, amounting in less than a decade to an exceedingly high, life-shortening cumulative dose.

The energy department recognizes that these doses exceed reasonable standards for public health protection — hence the pressing need for deeper analysis and a search for a more promising strategy.

It is likely that a license application showing a dose that is in compliance over a 10,000-year regulatory period, but that indicates significantly increasing doses beyond that time, will be subject to legal challenge even if considered acceptable by NRC. A repository design that could avoid this difficulty, if such a design is feasible, should be given serious consideration.

Engineering design considerations will differ depending on the period of concern. The pre-closure period, although considerable, is comparable to the usual time for which engineered structures (e.g., bridges, tunnels) are designed to perform. Primary concern will likely be occupational exposure of workers involved in construction and maintenance of the open repository and its contents.

As noted earlier, the much longer postclosure period (to 10,000 years and beyond) requires a less traditional engineering design approach. However, it is worth recalling that the decision to use underground *(geological)* settings for waste repositories was made, at least in part, because rock is a natural material that is known to have existed in stable form for *many millions* of years. Prediction of performance for a small fraction of this time into the future involves much less uncertainty than is the case for fabricated materials that have been available on the order of 100 years only. (The Swedish [SKB] decision to select copper as their waste-package material was based in large part on the fact that native copper deposits are known to have survived for millions of years in groundwater environments similar to those proposed for their waste repository.)

Primary Attributes of a Yucca Mountain Repository Design

DOE's viability assessment (VA) lists the following four main attributes of a repository at Yucca Mountain that can influence the release of radionuclides to the biosphere:

- water contacting the waste package;
- waste-package lifetime;
- mobilization rate of radionuclides; and
- concentration of radionuclides in water.

These attributes serve as primary guides for DOE in establishing its repository safety strategy (RSS). *Each attribute has been further subdivided into principal factors of the so-called reference design.* Alternative design features have also been defined as possible contributors to an enhanced design (i.e., to improve the overall safety of the repository). The inter-relationships among these elements are all contributors to the RSS (see VA, Vol. 2, Table 8-3, p. 8-5).

Clearly, if water percolation into the waste-filled drifts could be avoided (i.e., if no water contacted any waste package), then the remaining three attributes become of little or no significance. All are dependent, in large measure, on contact of the groundwater with the waste package.

As noted by Shoesmith and Kolar (1998) in summarizing their study of the corrosion resistance of metallic alloys and the possibility of long-lived waste packages:

> *If the contact of seepage drips with the waste package is avoided, then extremely long lifetimes, in excess of 10^6 years, are predicted. This would suggest that the adoption of any engineering option to avoid contact between drips and waste packages would be a good idea.*

Given the potential benefits of elimination of water contact with the waste package, it is surprising that little consideration has been given in the VA to:

(1) diversion of inflowing water *before* it reaches the repository horizon, and

(2) use of a multi-level design (i.e., to reduce the repository plan area, or *footprint*, in order to minimize the potential for dripping into the drifts.

If, as appears to be the case at Yucca Mountain, flow through the unsaturated zone is predominantly vertical, at least in the southern portion of the proposed repository location, then elimination, or at least major reduction, of infiltration to the drifts seems technically feasible.

If net infiltration could be eliminated, major TSPA uncertainties would be removed, and doses would be reduced dramatically, especially beyond 10,000 years.

Elimination of Water Infiltration

The following two engineering options are within current technology and offer the possibility of eliminating water inflow to the repository:

(1) Surface modification (i.e., engineered fill), and

(2) Underground repository infiltration shield.

Surface modification is mentioned briefly in DOE's viability assessment (VA, Vol. 2, Sec. 8.2.2, p. 8-7). The repository shield concept is not considered.

Surface Modification

> *Net infiltration into the mountain could be significantly decreased if the surface of the mountain were modified . . . . Likewise, facilities for drainage of water to enhance runoff could be designed. Because* ***these effects could potentially eliminate net infiltration*** *at the site, the potential importance to performance could be high* (VA, Vol. 2, p. 8-7, emphasis added).

Standard procedures of surface mining and site rehabilitation could be used to cover the repository site with an impermeable cap and drainage. As noted in the viability assessment:

> *Surface modifications and near-field rock treatment can be independently evaluated* [i.e., without affecting other features of the design] *so this alternative concept was not retained for further consideration as an alternative design concept. However, the merits of these features will be evaluated in a separate study (VA, Vol.2, Sec 8.2.4.2, p. 8-12).*

Surface modification treatments (e.g., several meters of thickness of an impermeable barrier, such as clay, overlain by a drainage layer of large river gravel covered by, say, 10—15 m of alluvium) are well within current surface mining technology. However, the surface topography above the proposed repository is variable, so that this surface treatment could be costly and environmentally objectionable.

One of the potential shortcomings of surface modifications alluded to in the viability assessment (VA, Vol. 2, Table 8.5, p. 8-30), is the questionable longevity of such a barrier, due to erosion. However, erosion rates at Yucca Mountain are estimated (VA, Vol. 1, p. 2-26) to be less than 1.1 cm per 1000 years, or 11 m in 1 million years. DOE has given preliminary consideration to a more limited treatment of the surface, including a cover of alluvium over the existing surface (E. L. Hardin, personal communication, 1999), but this has not been pursued to date. Lack of permanence of the cover was one of the concerns cited.

Underground Repository Infiltration Shield, with Multi-Level Repository

An underground infiltration shield is particularly well suited to a repository in the unsaturated zone in fractured rock. where groundwater flow is predominantly vertical and the rock mass is anisotropic, both hydrologically and mechanically. At Yucca Mountain, fracturing (subvertical) is such that the vertical hydraulic conductivity is significantly larger than the horizontal conductivity. Similarly, the modulus of deformation of the rock mass is larger in the vertical direction than in the horizontal direction.

The infiltration shield concept is illustrated in Figure 2. In the example shown, the repository is laid out as a three-level system.[5] This alone, by reducing the plan area *(footprint)* of the

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[5]Note that this would also reduce the probability of penetration of a vertical igneous dike intrusion by a similar factor, e.g., from a probability of 1×10^{-7}/yr as currently estimated by NRC scientists to 3.3×10^{-8}/yr.

repository to one-third of a single-level design, reduces the exposure of the drifts to vertical infiltration by a factor of three. Although the shield principle can be applied to a single-level repository design, it is obviously more cost effective to use a multi-level design.

A numerical analysis of the effect of placing a fourth row of drifts (left open, for example, as ventilated observation and performance confirmation drifts (see VA, Vol. 2, p. 4-45) above the three repository levels was carried out by Professor Pierre Perrochet, University of Neuchatel, Switzerland, using the numerical (hydrological) code FEFLOW. The analysis, with assumptions and results, is outlined in Appendix I to this paper.

A single typical column of drifts was analyzed. This corresponded to the central column shown in the upper diagram in Figure 2, but with the upper slot replaced by a circular drift (see diagram in Appendix I). The flow conditions and rock mass properties were considered to be representative of those in the unsaturated zone at Yucca Mountain. It was assumed that the rock mass could be considered to behave as an anisotropic continuum (i.e., discrete fractures were not considered). A uniform vertical infiltration of 50 mm/yr (1096 x 10^{-5} m^3/d over the 80 m^2 potential capture area (per meter of drift) was assumed to occur 30 m above the top row of drifts. A wide range of hydraulic anisotropy was examined. For all anisotropies considered, at least 94% of the top infiltration bypasses the lower three (rows of) drifts. The fluid pressure head above the lower drifts is reduced because of the proximity of the overlying drift, thus enhancing the potential for diversion of water around the lower drifts.

This calculation can be criticized in that it assumes the drifts to be circular and smooth, thus enhancing flow deviation around the drifts — as indicated in Figure 3 (after Philip et al., 1989; Philip, 1990). The presence of discrete fractures in the roof would increase the potential for water to drip into the drifts compared to the case analyzed — viz. that of a smooth opening in a continuum.

This criticism can be circumvented if the upper drift is replaced by a slot, say, 2 m high and ~10 m—20 m wide. Each such slot could be inclined slightly, as shown in Figure 2, and backfilled so as to establish a *flow diversion barrier*, to ensure that any infiltration from above the slot would drain into the rock mass outside the perimeter of the repository. Excavation of the 15 m—20 m slot would serve a dual purpose. A zone of enhanced fracturing would tend to develop above the slot (this would be further enhanced during the thermal cycle after the repository is filled with waste.) Any water infiltrating into the zone would drain into the slot; any remaining flow would be directed into the rock mass away from the drifts. Thus, both mechanisms (capillary diversion around and fracture flow into the slot) act to prevent flow into the drifts.

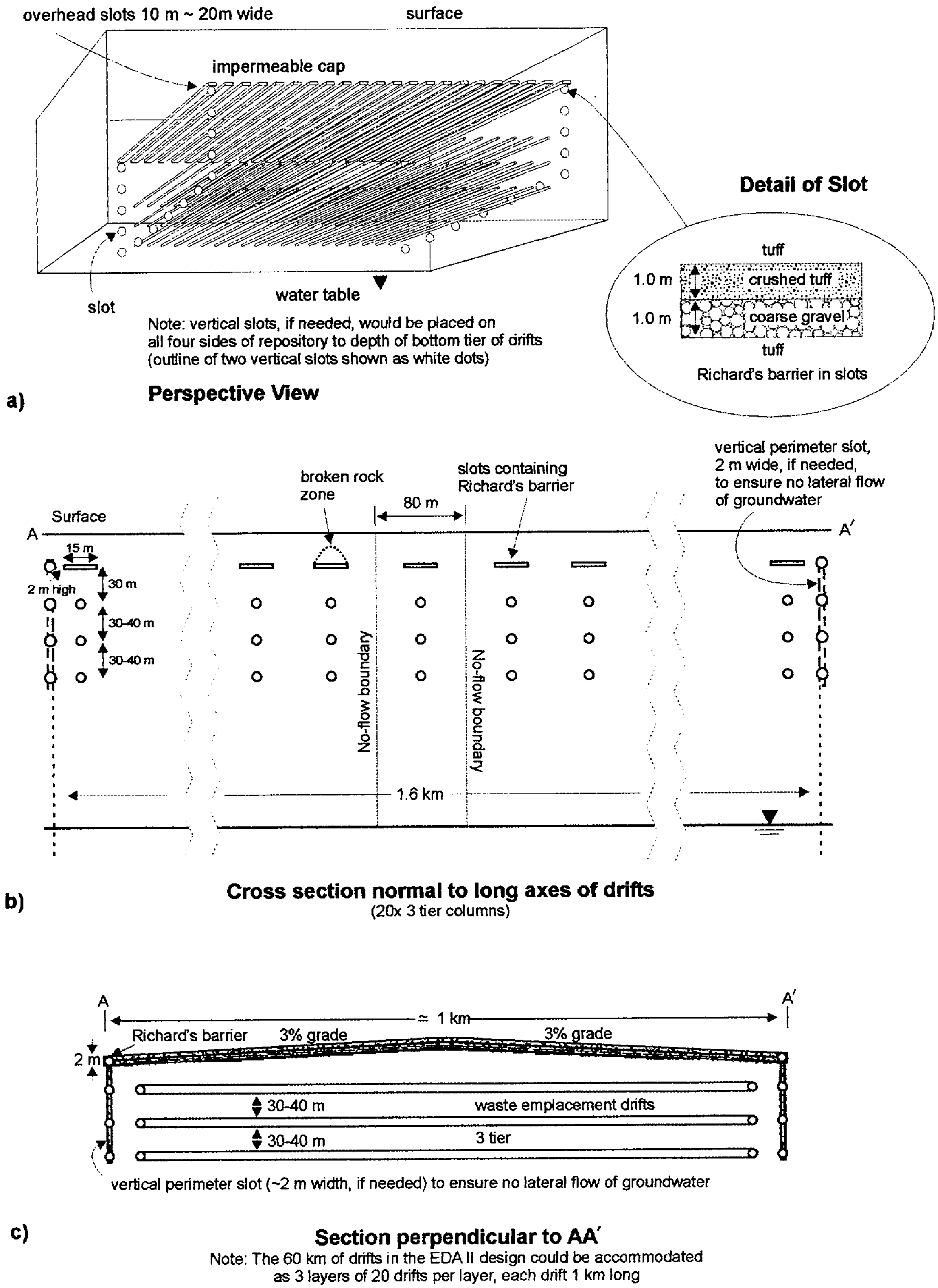


Figure 2. Underground Repository Infiltration Shield

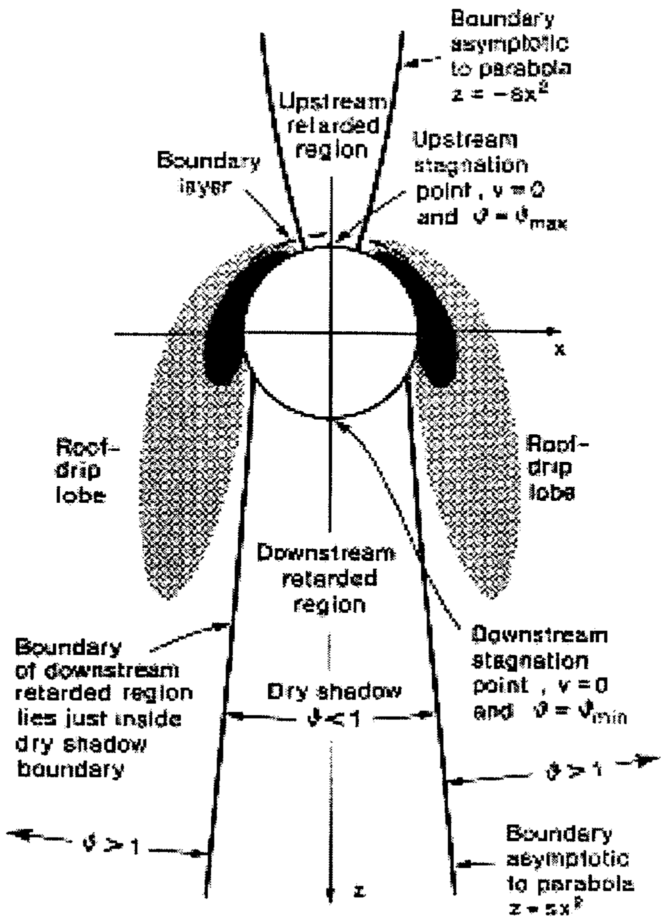


Figure 3. ***Seepage around cylindrical cavities (schematic diagram illustrating critical points and regions of the flow field)***

Richards Barrier

This flow-diversion system incorporates two layers of material with contrasting hydraulic conductivities — a fine-grained porous layer overlying a coarser-grained layer, also porous (see EPRI (1996), pp. 1—2 et seq. for details). The capillary pressure established within the pore space in the upper layer material at the interface with the lower layer acts to prevent flow into the lower layer and promote flow laterally in the upper layer. Currently, the DOE is engaged in considerable study of the Richards barrier. The intention is to cover the waste packages with a "tailored backfill" possibly designed as a Richards barrier to divert water drips from the roof of the drift away from the packages (see Figure 4)[6]. Figure 2 shows a similar two-layer arrangement of backfill for the slots in the proposed repository shield.

[6]It may be that the behavior of a Richards barrier over very long times (i.e., 10,000 years and longer) could be considered doubtful. It is believed that a simple drain, consisting of graded, more or lessuniformly sized granite boulder (river gravel) would suffice to establish free draining of the slots.

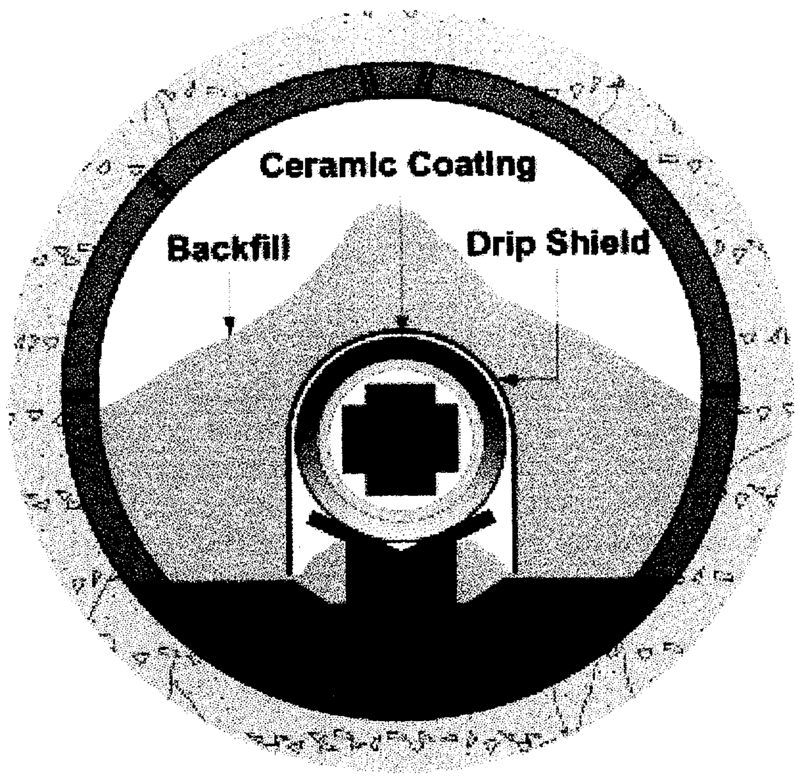


***Figure 4.* Near-Field Engineering Measures to Prevent Dripping on to Waste Packages (It is planned to place the backfill in two layers as a Richards Barrier, with fine-grained rock material overlying a coarser-grained rock material.)**

Potential for Lateral Flow at the Repository Horizon

The repository shield design described above is designed to be effective against vertical infiltration. It will fail if there is significant lateral flow across the repository. Lateral flow is possible, and is known to occur both above and below the proposed repository horizon. Within the proposed horizon (particularly, the southern region), flow appears to be dominantly vertical. As noted in the DOE viability assessment (VA, Vol. 1, p. 2-38),

> *. . . evidence indicates that surface infiltration generally moves downward rapidly in fractures through the Tiva Canyon tuff until it encounters the non-welded Paintbrush tuff. Flow in the non-welded unit appears to be predominantly in the rock matrix although fast flow paths along faults, fractures and other high permeability zones are present locally. In general, it appears that the Paintbrush non-welded unit attenuates (slows) and distributes flow downward, perhaps for periods of up to thousands of years. After migrating through the Paintbrush tuff, water moves into the welded Topopah Spring tuff* [Note: The proposed repository horizon is in the Topopah Springs formations] *where flow again appears to be dominantly in the fractures. The distribution of flow is heterogeneous; in some areas characterized by widely dispersed or poorly connected fracture systems, percolation fluxes may be very low. In areas with highly transmissive features*

such as faults or dense fracture networks, significant volumes of water may move downward rapidly.

This discussion suggests that lateral flow across the repository is likely to be minimal, so that (horizontal) slots above the waste-filled drifts will eliminate most, if not all, of the potential infiltration into the repository. It is entirely feasible technically, if deemed advisable to further reduce uncertainty, to construct a vertical perimeter shield around the entire repository, as shown in Figure 2. This would require a single vertical column of four 5-m-diameter drifts, located on the same level as the repository drifts and slots, along each side of the repository periphery. A narrow vertical zone of enhanced permeability could then established by blasting, using the VCR (vertical crater retreat) method (or a similar stopping procedure). Blasting would be conducted in vertical holes drilled downward from each overlying drift. The blasted rock would fill the underlying drift such that little, if any, of the broken rock would need to be removed. The aim is to establish a highly transmissive vertical flow pathway around the periphery of the repository; it is not necessary or desirable to create a vertical excavation. Alternate, less expensive techniques (e.g., creation and propping of hydraulic fractures from vertical holes along the drifts) could also be considered.

The horizontal slots and perimeter drifts could be used for monitoring (e.g., by microseismic and other geophysical techniques) repository performance during the preclosure period and beyond, if necessary. Since these openings would be ventilated during this period, any infiltration would be carried out as vapor in the air stream.

Additional Excavation Required for the Repository Shield

Horizontal Slots Only — The total excavation to develop 20 m-wide x 2 m-high slots would be the equivalent of 40 km of 5 m-diameter drifts. The EDA II repository design envisages a total of 60 km of waste-filled drifts. Thus, addition of the 20 m excavation slots would result in a total excavated volume less than the 110 km of drift excavation contemplated in the VA repository design.

"Full" Shield — The four drifts along the entire repository perimeter, if needed, would add a further 21 km of excavation (i.e., 4 x 2(1.6 + 1.0) km). It may be possible, in view of the reduced concern over reflux pathways between the (columns of) drifts, to reduce the spacing between drifts (currently 81 m). This would reduce the extent of the repository footprint plus the cost of generating the high-permeability vertical fracture zone between the drifts.

However, it is considered unlikely that construction of these vertical high-permability zones will be needed provided the repository horizon is selected appropriately, i.e., where the two sub-vertical joint sets are both well developed. They are orthogonal to each other, thus forming an effective barrier to lateral flow across the repository.

The preceding discussion suggests that it is technically feasible to ensure that essentially no infiltration into the repository ever occurs, for a cost that would not significantly exceed that of the VA repository design. This does not consider the added cost of a three -level repository compared to the VA single-level repository. DOE has considered a two-tier or split-level repository option, but did not examine the potential for water diversion. An increased cost of construction of 19% compared to the VA reference design was indicated (CWRMS/M&O Report

Design Feature Evaluation #25, Repository Horizon Elevation, April 2, 1999). It is also worth noting that the repository shield requires no reliance on the long-term performance of manmade materials. It should be relatively easy to establish the very long-time reliability of the repository shield.

The distinct possibility that the repository shield concept could reduce drift infiltration sufficiently to make the titanium drip shield (Figure 4) unnecessary — for a cost saving of $4.6 billion — strongly suggests that the repository shield concept deserves detailed study by DOE. Such a study should examine the implications of the multi-level arrangement (with overlying slots) on the optimum repository design.

Location of a Multi-level Repository at Yucca Mountain

A three-tier repository, as shown in Figure 2, would occupy a vertical interval of approximately 60 m~80 m in the Topopah Springs formation. Since the horizon proposed currently for the single-level repository is approximately at elevation 1080 m it appears that a three-tier interval from 1,040 m to 1,120 m in the central third of the current repository (see VA, Vol. 2, Fig. 4.21, p. 4-40) will remain well within the "groundwater surface plus 100 m" lower limit and within the "200 m cover" upper limit. The slot horizon would be some 30 m or so above the upper row of drifts, but this too will have almost 200 m of rock cover. Since the slot would contain no waste, a cover slightly less than 200 m is considered adequate.

Optimum Repository Layout

The VA reference design was a "hot repository" in which rock temperatures in excess of 200 °C were envisaged. A main intent was to prevent access of liquid water to the waste packages, at least for much of the regulatory period. Concern over the uncertainties associated with two-phase fluid flow behavior in the near-field of the repository and associated complexity of coupled (thermo-hydrological-mechanical-chemical) effects, especially in the near-field around the drifts, led to calls to revise the design to one in which the rock temperature was lower, preferably below the boiling point of water for much of the duration of the thermal cycle. The EDA II "lower temperature" design responds to these concerns.

The two designs are compared in Table 1 (from the presentation "Current Status of Repository Design," by Daniel G. McKenzie III, to the Drift Stability Panel, April 13, 1999).

The EDA II design has some merits, but also some disadvantages. Although the lower temperature system may be simpler *(perhaps!)* for purposes of analysis of near-field fluid (liquid water and water vapor) movement, the possibility that the high-temperature design may inhibit access of liquid water to the drifts is a feature that should not be abandoned lightly. Center for Nuclear Waste Regulatory Anslysis (CNWRA) staff (R. Green, personal communications, 1999) suggests that some counter-current flow may occur, -whereby water vapor may ascend within a fracture while liquid water may descend into the drift via the same fracture. The importance of this possibility in the context of a repository shield design would need to be assessed.) Also, as noted in the EPRI report (EPRI, 1996, p. 1-2):

The proposed DOE schemes for lower thermal loadings would not eliminate completely any of the coupled thermal effects causing concern at Yucca Mountain, although the proposed schemes would reduce the magnitude of at least some of these effects. For example, lowering peak temperatures below the boiling point does not eliminate the potential for evaporation of liquid water from the rock followed by buoyant convection and subsequent condensation farther afield. In order to reduce dramatically thermal effects in the very near field around the containers, the amount of spent fuel contained in an individual container would have to be dramatically reduced or the decay time of the spent fuel would have to be significantly extended (well beyond 100 years). Neither of these approaches seems so practical since both would dramatically increase disposal costs.

Table 1. Comparison Between the EDA II and VA Repository Design Options

EDA II Design	DOE VA Design
60 MTU/acre	85 MTU/acre
1,050 acre-layout	741 acre-layout
60,000 m of emplacement drifting for statutory waste capacity	117,000 m of emplacement drifting for statutory waste capacity
2-5 m^3 /s/drift airflow	0.1 m^3/s/drift airflow
81 m drift spacing	28 m drift spacing
Line load	Point load (3 m between packages)

It is instructive, in this regard, to consider the performance of a multi-level EDA II design, as illustrated in Figure 2. The switch to a "line load" of waste packages (i.e., with the packages placed essentially adjacent to each other along the drift) compared to a "point load" (packages separated by several meters along the drift) and a much increased spacing between drifts (81 m for EDA II; 28 m for the VA Reference Design), together with some (low) velocity ventilation of the EDA II drifts, was intended to simplify the convective flow paths with reflux via the cool region in the center of each pillar.

With the multi-level design, the rock temperatures are likely to be increased, principally along the vertical axis between the drifts. The region between the pillars will be less affected, although raised somewhat. Convection cells of heated water and water vapor would form, driving the fluids upward into the slots, where it would tend to condense on the coarser rock in the lower portion of the Richards barrier, flowing along the inclined drifts to drain outside the repository. Continued heating would eventually dry out the rock between each column of drifts. This pathway provided by the slots would tend to eliminate the need for a pathway for the condensed reflux between the pillars, although concentration of the overburden stress through the pillars would induce a small tension tangential to the central vertical axis of the pillar, thereby tending to open the reflux pathway. In this regard, it should be noted that the intensity of the vertical stress concentrations in the pillars will persist to a greater depth than in the case of isotropic and unjointed rock (i.e., the "aperture opening" effect may be more significant in the jointed rock (see Goodman, 1989, Figs. 9.10 and 9.11 pp. 352—361). Shears induced at the corners of the slots could also cause fracture dilation, especially during thermal cycles.

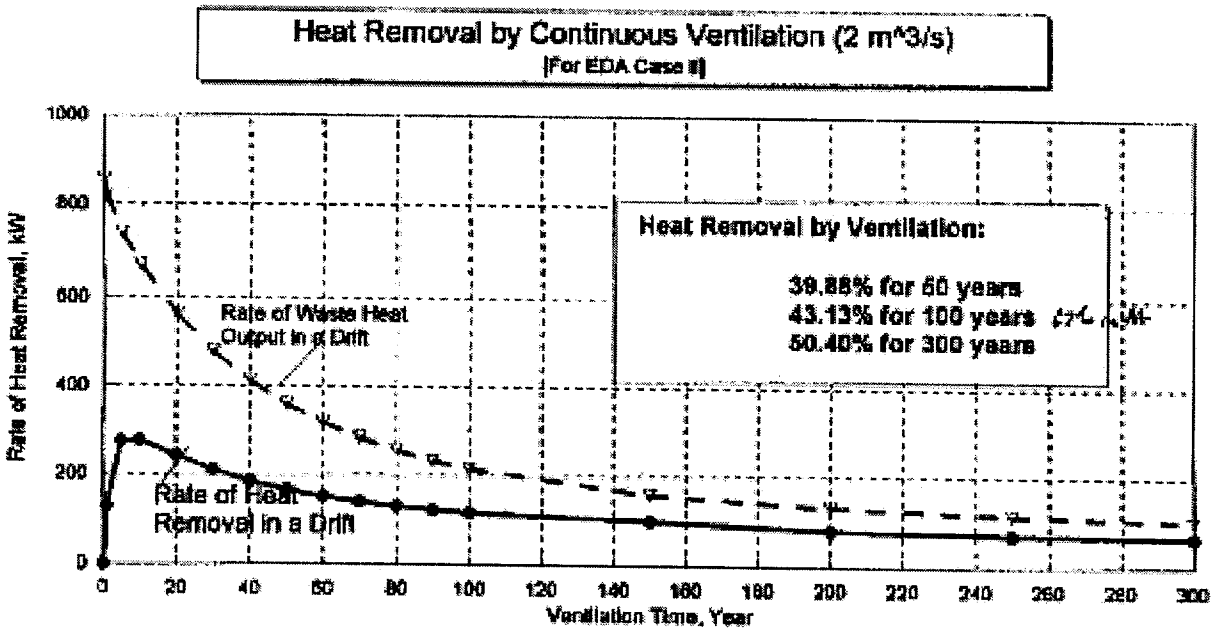
It should be possible to reduce the 80-m drift spacing of EDA II somewhat (say, to 50 m). This would increase the temperature along the center-pillar axis, but the stress concentration in the now narrower pillar between the slots would increase, which may increase shear and dilation of fractures. The reduced pillar size would reduce the plan area of the repository, thereby either reducing the extent of any vertical perimeter shield or increasing the capacity of the repository. Chemical dissolution of minerals species (e.g., silicates) in the rock by the hotter fluids in the near field, with condensation upon reaching the slots would tend to develop a low-permeability "skin" along the slot floor during the thermal period. This would be beneficial to drainage of condensate along the drift.

Obviously, more detailed analysis and optimization studies are needed to establish the merits of the multi-level design with the repository shield in order to establish the merits of this concept vis-à-vis the proposed single-level designs.

Control of Repository Temperature

Reference has already been made to the perceived benefits of reduced repository temperatures in order to simplify the near-field fluid flow regime. Low temperatures are also desirable to reduce corrosion of the waste packages. The EDA II waste package involves a 2-cm-thick outer cylinder of C-22 alloy steel, with a 5-cm-thick inner cylinder of stainless steel (316NG).

Shoesmith and Kolar (1998) argue that pitting and crevice corrosion of C-22 are unlikely to occur at temperatures below 150 °C and 102 °C, respectively. The authors present detailed discussion of the corrosion processes, but conclude that a conservative design limit is to take 80 °C as the temperature below which crevice corrosion of C-22 can not occur (Shoesmith and Kolar, 1998, p. 5-8, para. 1). Also, it is noted that water must be present for significant waste package corrosion to occur. A relative humidity less than 70% and a temperature below 80 °C are sufficient to reduce the possibilities of corrosion of the C-22 alloy to insignificant values, i.e., yielding estimates of waste-package lifetimes considerably longer than the 10,000 years of the regulatory period.



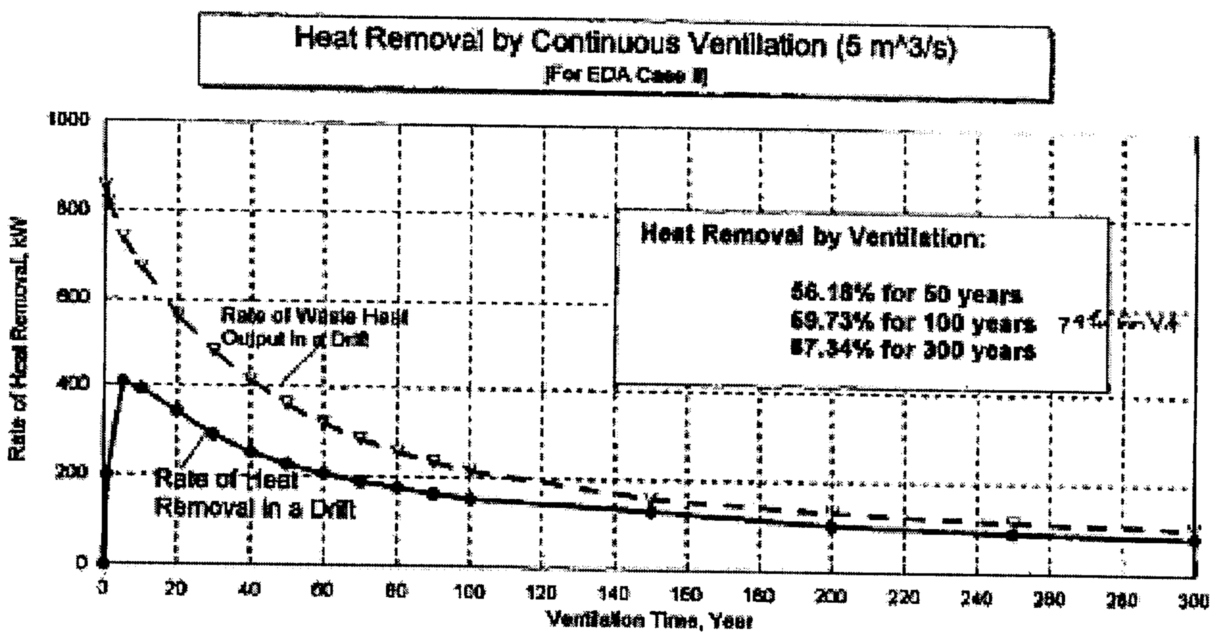


Figure 5. Heat removed by continuous ventilation of waste-filled drifts during the pre-closure period (2m^3/s air flow in a 5-m diameter drift corresponds to an air velocity of 20 ft/min, or 0.23 mph)

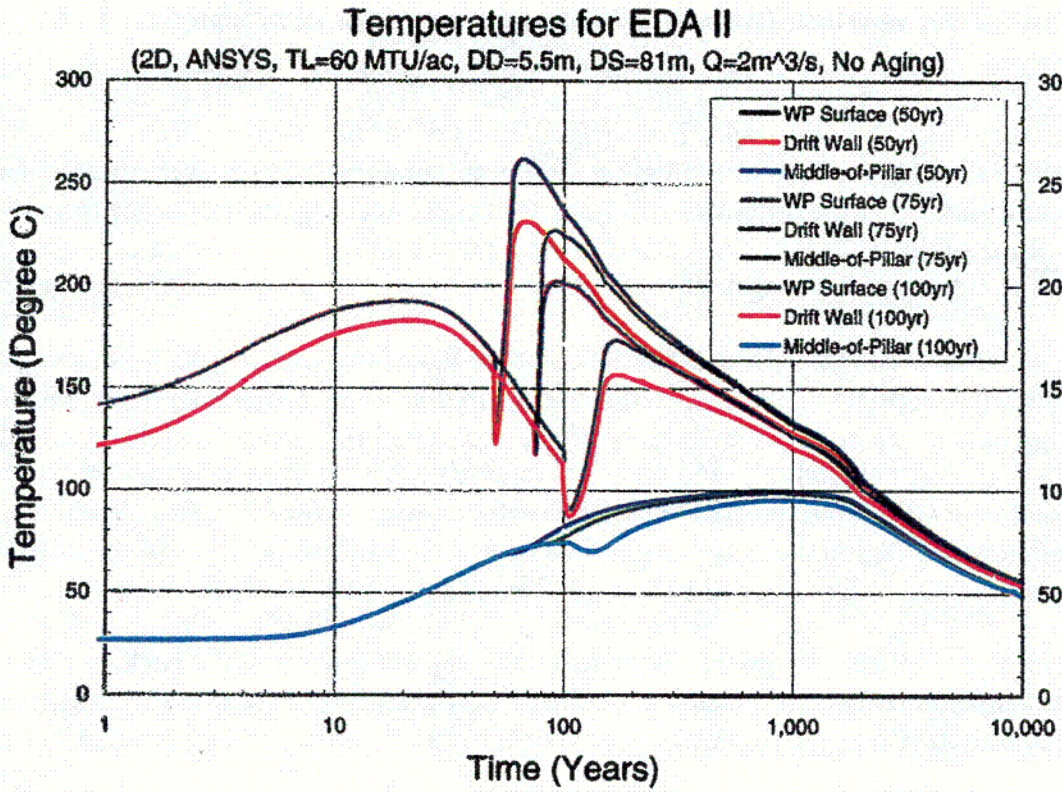


Figure 6. ***Effect of backfill on the evolution of temperature in the repository drifts (EDA II design)***

Figure 5 (kindly provided by the DOE, courtesy of R. Craun, 1999) indicates that drift ventilation can remove significant quantities of heat from the waste packages, especially during the first 50~100 years, when heat generation is most intense. Figure 6 (courtesy of D.G. McKenzie, April 1999) indicates that, with the EDA II design, drift ventilation of 2 m^3/s, and no aging of the waste:

(1) The waste package surface and the drift wall both exceed 150 °C for several years after installation, and

(2) Active ventilation of the drifts, either natural or forced, can also reduce the humidity. Stellavato and Montazer (1996, pp. 25—26) have used the atmospheric/hydrologic code ATOUGH to model heat removal from a ventilated repository.

They advocate design of the repository to allow air to flow continuously and indefinitely through the waste-filled drifts driven by natural ventilation. In their report, the authors conclude that:

> *By considering a naturally ventilated repository (after construction) and taking advantage of the thermal drive of the waste package, the repository may be kept dry during at least the first 10,000 years if not longer. The amount of moisture removed from the rocks during this time will create a thick low-saturation skin*

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around the drifts that will require thousands of years to re-saturate. Ventilation can also remove large amounts of heat generated by the waste canisters.

The authors' analysis indicates that the rock temperature never exceeds 25 °C during the ventilation period. The topography and surface layout of the proposed repository at Yucca Mountain is favorable to natural ventilation (and ventilation produced by waste heat generation), but it seems likely that the drifts will collapse over time, increasing the resistance to ventilation. Partial filling of the drifts with "moderately large" boulders to ensure some air access to the packages could be considered, but the resultant overall resistance to flow would be considerable.

Clearly, there is merit in preclosure ventilation of the repository with respect to limiting temperatures. Ventilation also tends to develop a "dryout" zone in the rock. Measurements over the past several years suggest that a region of approximately 100-mm radial thickness is dried out annually. Although the radial extent may not increase linearly with time, it appears that a region not greater than 10 m from the drift excavation will be "dried" over 100 years. With interruption of ventilation, this region will resaturate, probably at a comparable rate, so that the drift will be resaturated (i.e., partially) after the order of 200 years from installation of the waste. Thus, for almost all of the 10,000 years of the regulatory period, the waste packages (and backfill?) would be subject to a humid environment. With the C-22 alloy outer cover of the packages, and a package temperature not significantly above 100 °C, the alloy will corrode very slowly, if at all. This resaturation rate would be slowed considerably if the repository shield concept was used.

The preceding calculations suggest that, if one would hold the temperature of the C-22 waste package below 80 °C , some combination of waste form "blending" in the drifts, aging of the waste in surface facilities before emplacement in the repository, and active *vigorous* ventilation of the packages for at least 50—100 years may be necessary in open drifts. An *open drift* implies that the waste package will not be covered. —i.e., the waste package surfaces should be accessible to the ventilation. Tailored or "getter" backfill in the drift invert below the waste package could still be used.

Design considerations such as those outlined above suggest that it is entirely possible to engineer the natural setting of the unsaturated zone at Yucca Mountain to ensure that a high-level waste repository will be demonstrably safe for an indefinite period into the future. The *umbrella principle* of the repository shield is simple and can be comprehended easily by the general public.

Drift Stability

It is planned to locate the repository in the Topopah Springs tuff formations. For purposes of drift support/reinforcement and stability analyses, the formations can be divided into two general categories:

(1) *Non-lithophysal tuff.* - These formations contain three relatively well-developed joint sets. (Two are subvertical: joint set No. 1 has a dip of 77° and a dip direction of 40° ; joint set No. 2 has a dip of 80° and a dip direction of 130°. One is sub-horizontal: joint set No. 3 has a dip of 25° and a dip direction of 300°); and

(2) ***Lithophysal tuff.*** -These formations contain three-dimensional voids — approximating spheres or ellipsoids in most cases — or *lithophysae* generated as gas pockets during the

period of deposition of the volcanic tuff. Some of the lithophysae can approach 0.5 m in diameter, although most are smaller (predominantly 7—15 cm in diameter). Also, fractures in the lithophysal rock are shorter and less persistent than in the other units, and often terminate (or originate?) at the lithophysae.

> It seems likely that the lithopyhsal zones will be stronger and stiffer (i.e., higher rock mass modulus) than the non-lithophysal zones because of the lesser influence of through-going joints. The higher modulus would result in higher thermally induced stresses for a given temperature, so that the extent of *damage* during the thermal cycle could be comparable for both lithophysal and non-lithophysal tuffs.

It seems to the writer that excavations with rock reinforcement should be stable in both formations. The following discussion will examine the likely mechanical response of the two types of formation to loads generated in a repository. The stability of the repository drifts is of particular importance for the preclosure period, and can have consequences for the long-term performance of the repository, especially if the drifts are not backfilled.

Preclosure Stability

Although there is a wealth of experience in designing and constructing tunnels of the general dimensions of the repository drifts, and there are examples of tunnels that have remained stable for much longer than 100—300 years, design of a repository is unique in that a major thermal cycle is involved. For the case of a hot repository, this heating imposes substantial additional stresses on the rock and any rock lining. The likelihood that a concrete lining would be seriously and adversely affected by the high temperatures is — in part, at least — the reason why an Expert Panel on Drift Stability has recently recommended the use of rock bolts and wire mesh as being a more suitable support system than a concrete liner.

Postclosure Stability

DOE lists the following information needed with respect to performance assessment (PA) for ground support/drift stability (R. Howard, Yucca Mountain Drift Stability Panel, April 13, 1999):

Ground Support/Drift Stability Information Needs for PA (FEPs)[7]

- masses and spatial distribution of ground support materials
- nature and rates of continuous degradation processes
- nature and probability of disruption by rock fall
- nature and probability of disruption by seismic motion

Of these, the first can be answered as soon as a support system is selected. The remaining three require an understanding of the long-term, time-dependent behavior of the rock mass ***only if the drifts are not backfilled***. If the drifts are backfilled, then these issues are no longer of concern.

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[7] FEPs are features, events, and processes that are considered to influence repository performance.

No firm decision has yet been made concerning whether to backfill the drifts after waste emplacement.

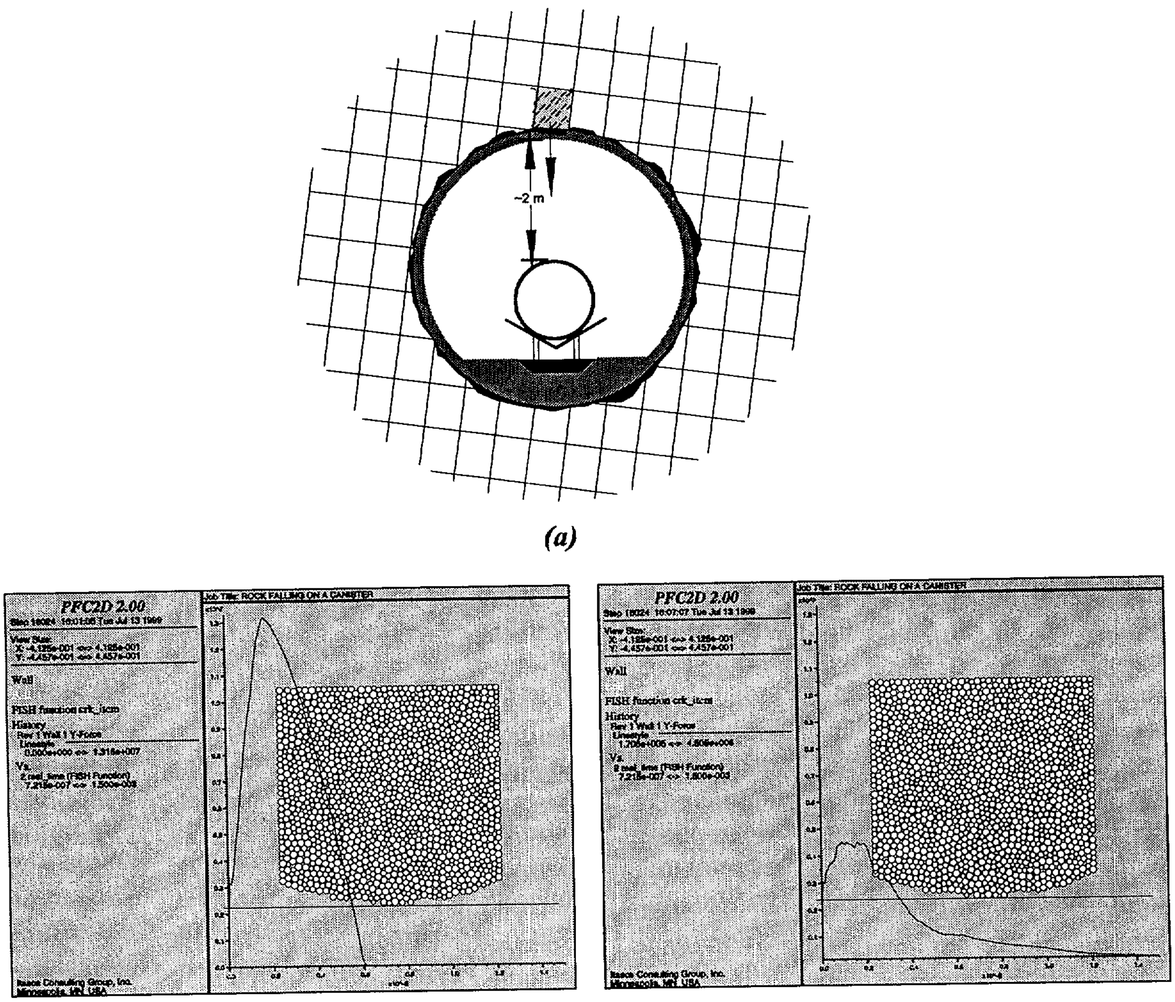
Numerical (discrete element) models currently in use to assess drift stability at Yucca Mountain have a significant limitation in that the rock blocks in these models, although deformable, are assumed to have infinite strength (i.e., they cannot break). This results in significant over-estimation of the consequences of rock falls on to waste packages. Considerable improvement in prediction of both (1) the consequences of heating on spalling of the drift walls and (2) the behavior of falling blocks can be obtained using a code such as the micro-mechanics numerical code PFC (Potyondy and Cundall, 1999) that allows the blocks to break under applied loading. Some indication of the difference that may be expected is demonstrated by the simple example of a rock block falling 2 m from the roof of the drift onto a waste package, as shown in Figure 7. The resultant force-versus-time history during the impact is shown in Figures 7(b) and 7(c) for the two cases in which (b) the block has infinite strength, and (c) a similar block has the (finite) strength of Yucca Mountain tuff. Fragmentation of the block (Figure 7(c)) traps a substantial proportion of the kinetic energy and momentum of the block with the result that, in this case, the peak force on the waste package is reduced to approximately one-third of the value indicated with the infinitely strong block.

Thermal loading and seismic effects can be considered in the PFC code. The rate of degradation over a long time can also be estimated, but this would require laboratory data on the strength of tuff (and joints in tuff) as a function of applied loading conditions (and possibly thermal conditions). Such data may not be available. The *pattern* of collapse with time can be examined for various *assumed* strength-degradation models. If this indicates that the pattern is relatively independent of rate of degradation, knowledge of the degradation pattern may suffice for PA purposes. Another approximate approach is to assume that the joint cohesion declines progressively in time toward zero. Frictional properties may decline somewhat, but are likely to remain significant.

It is anticipated that an analysis using PFC would indicate progressive spalling of the drift wall and collapse of *relatively small* rock blocks on to the packages. This would further reduce the severity of any rockfalls on to the waste packages.

Time-dependent deterioration of rock strength (and possible collapse) can occur whenever rock is loaded in compression beyond 40% to 50% of its ultimate compression strength. Stresses significantly above this level could be generated in the rock during the thermal pulse period of repository operation. (In the case of Yucca Mountain, the stress induced in the rock by temperature increase is approximately 0.5MPa/°C for an assumed modulus of deformation of the rock mass of $E = 6$ GPa.)

Recognition of the limited value of classical geotechnical engineering design approaches in prediction of rock mass behavior for repository design has stimulated studies to obtain a more fundamental understanding of the physical principles that control time-dependent failure in rock. The report by Potyondy and Cundall (1999), describing studies being conducted for the Canadian nuclear-waste isolation program (and including the influence of heat in degrading rock strength with time) outlines valuable developments on this topic.



(a)

(b) *(c)*

Figure 7. ***Effect of finite rock strength on the impulse generated by free fall of a rock block onto a waste package (PFC model) (The block in Fig. 6(c) has the same deformability as the (infinite strength) block of Fig. 6(b), but a strength corresponding to that of Yucca Mountain tuff)***

Effect of Heating on Drift Stability

Figure 8 illustrates the change in stresses produced in the periphery of an unsupported drift as the result of heating, in this case to 145 °C, assuming that the rock mass has properties almost equal to those of intact tuff (i.e., RMQ 5). The initial insitu stresses were assumed to be approximately 10 MPa vertical. (This is equivalent to a depth approaching 400 m and 3 MPa horizontally). Under these stress conditions, the tangential stresses around the drift preceding heating would reach a maximum compression of approximately 26 MPa acting vertically across the central horizontal axis. Assuming a rock mass modulus of 32 GPa (i.e., RMQ5 rock properties), the effect of heating to 145 °C is to add compression on the order of 120 MPa more or less uniformly around the tunnel wall if the rock retains the RMQ5 properties and remains elastic.

If the rock properties are *degraded* to those of rock of RMQ1 quality, the stresses shown by the solid lines in Figure 8 are developed. The effect of a total of 50 years of heating, after which high temperatures (and stresses) have penetrated further into the rock, is shown by the dotted stress distribution. This results in a zone of inelastic deformation such as indicated in Figures 8 and 9. It is seen that the stress distribution and extent of inelastic deformation depend heavily on the rock properties. Recent results of insitu modulus measurements in the heated drift experiment indicate that the rock mass modulus (of deformation) increases from the order of 6~7 GPa at ambient temperature to higher values at higher temperatures. This is due, very likely, to expansion of the rock and consequent closure of the rock joints with increase in rock temperature. It is unlikely that the rock mass modulus in the jointed rock will reach the laboratory value for intact rock (32 GPa). In the lithophysal tuff, however, the modulus can be expected to be higher than in the non-lithophysal jointed tuff.

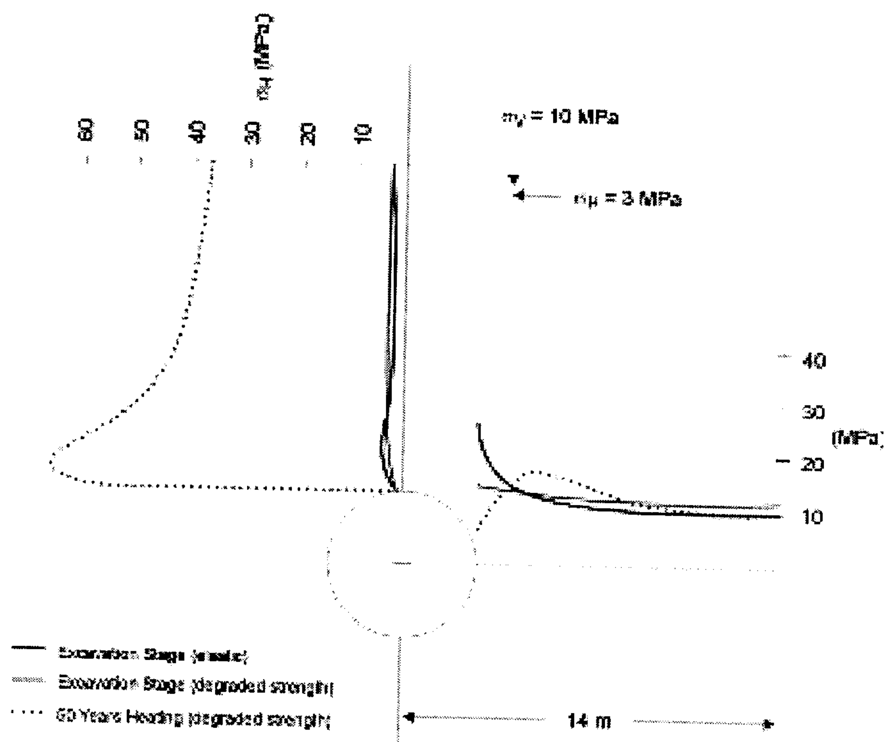


Figure 8. Effect of Heating on Stresses Around One of a Series of Excavations

Figure 9 shows the extent of joint slip that occurs before (Figure 9(a)) and after (Figure 9(b)) heating when a PMQ5-quality jointed rock mass is subject to heating as described for Figure 8.

Figures 9(c) and 9(d) show the results of numerical modeling in which a 5-m-diameter unsupported open drift is subjected to two identical seismic events, one that occurs before heating (Figure 8(c)); the other (Figure 8(d)) that occurs after 50 years of heating of the rock to a maximum temperature of 145 °C at the tunnel wall. The regions of joint slip are shown in Figures 9(a) and 9(b), and the rockfall due to the two seismic events in Figures 9(c)and 9(d). It is seen that the rock fall is considerably reduced for the heated rock. This is because the increased temperature superimposes a high compression all around the tunnel, tending to “clamp” the rock blocks together, and preventing fallout.

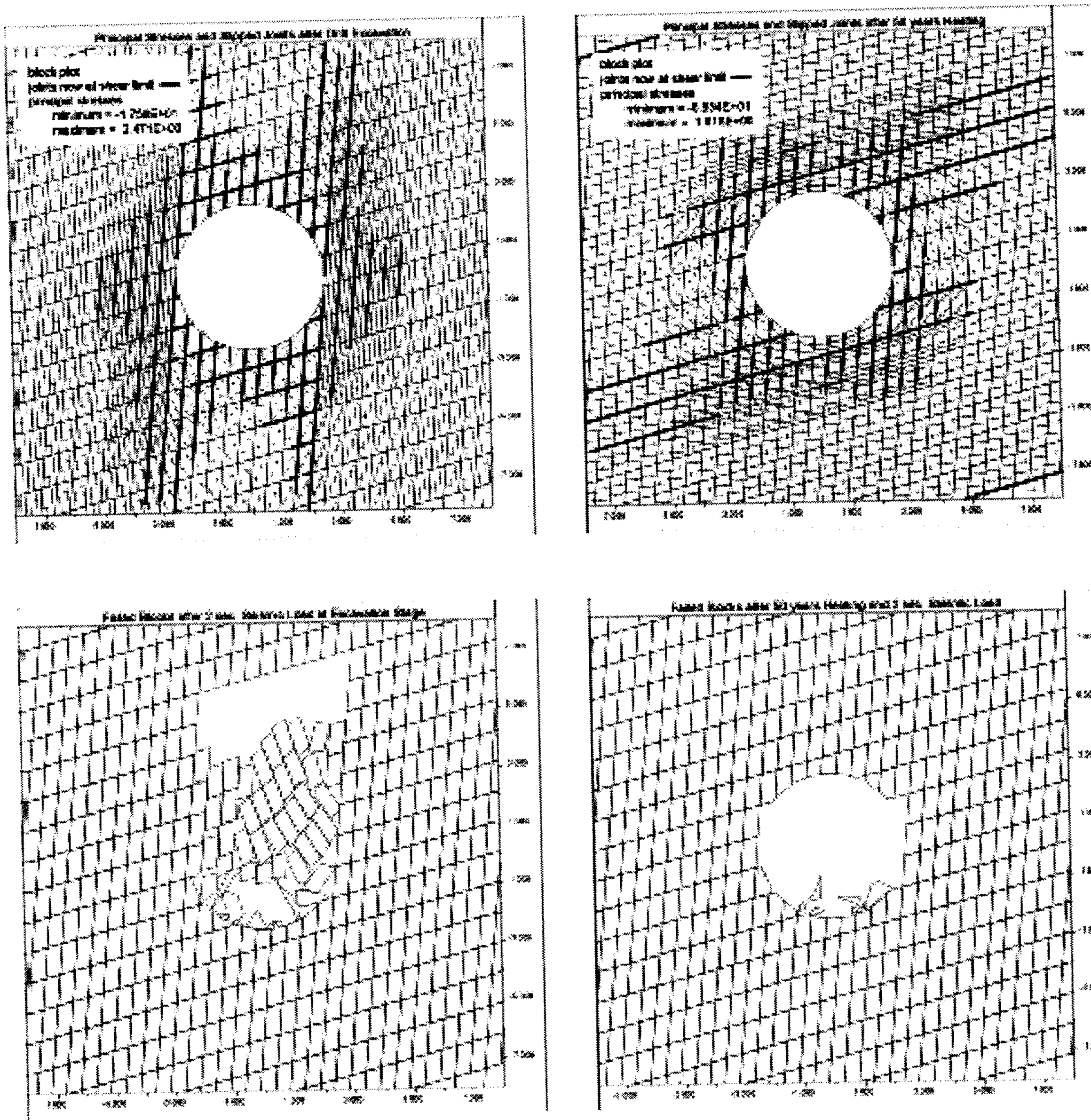


Figure 9. Effect of Heating (a,c) on Drift Stability and Seismic Event Before (b) and During (d) Heating

Thus, the consequences of a seismic event will depend very much on when the event occurs with respect to the thermal loading produced by the waste package. Upon cooling, the induced thermal stresses will disappear, and some additional collapse could occur. It was found, during the study of seismic effects mentioned above, that the second seismic event in each case caused little additional rockfall. However, the effect of time-dependent weakening of the rock mass was not considered. It seems probable that additional collapse would occur if this factor were added.

Figures 10(a)—10(d) show the results of numerical modeling to simulate various support options and assumed rock conditions.

The rock joints are assumed to have an initial (high) friction angle of 56° and a cohesion of 0.07 MPa. Other properties are those for RMQ5 rock (as defined by the M&O contractor). The reinforcement [grouted bolts (c)],or support [concrete (d)] is then installed, or the drift is left unsupported [(a), (b)] depending on the case considered. The rock is then heated to 100 °C. Joint slip and rock failure occur. Then, in order to simulate time-dependent degradation of the rock joints, the joint friction angle is reduced to 35°. Except for case (b), the joints are all assumed to be continuous. In case (c), the joints are *non-persistent*, consisting of alternate l-m-long segments of intact rock and joint, for which the friction angle is degraded to 35°.

Results indicate that the extent of the damage zone depends primarily on the frictional properties of the joints. Non-persistent joints (case (b)) behave essentially as intact rock, so that the extent of the damage zone is significantly reduced compared to that produced with continuous joints (case (a)); see the discussion of the lithophysal rock zone, below. Grouted rock bolts (case (c)) reduce considerably both the slippage on joints and the extent of the damaged region. Case (d) indicates that the elastic liner installed with a gap between the crown of the drift and the top of the liner to simulate a noncontinuous liner/rock contact does little to reduce the extent of damage compared to the case in which there is no support (case (a)), although the liner does, of course, prevent the rock fallout that would be very likely to occur in case (a).[8]

Lower Lithophysal Rock Zone

A brief analysis of the mechanical properties of lithophysal tuff (see Figure 11 and related discussion) suggests that the overall mechanical response to stresses (including thermal stresses) in these zones may be less influenced by joints and joint slip than is the case in the non-lithophysal zones. Thus, the rock mass strength in the lithophysal tuff may be somewhat higher, but the modulus of deformation will also be higher. Because the induced thermal stresses are directly related to this modulus, the *ratio* of stress:strength will change less. It seems, therefore, that from the mechanical stability perspective, drifts (e.g., for a multi-level repository) may be located in either or both lithophysal and non-lithophysal regions.

Both the Nuclear Waste Technical Review Board (NWTRB) and the NRC have criticized DOE for its failure to determine the insitu mechanical properties of the lower lithophysal rock, in which approximately 70% of the repository will be located. (Most of the rock properties have been determined for other, non-lithophysal units.)

An analysis was conducted to assess the influence of the lithophysae (assumed to be spheres) on the strength of the rock mass. Since, as noted in the discussion of Figure 10 case (b), non-persistent joints tend to exhibit the same strength as the intact rock in which they are found, the analysis assumed that the rock around the lithhophysae had the same properties as those defined by RMQ5. As stresses are increased (in this case, due to heating) on the rock, the lithophysae behave essentially as interior (spherical) excavations, i.e., stress concentrations occur around the

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[8] These analyses were made available, courtesy of Dr. R. Hart of Itasca Consulting Group Inc. Dr. Hart is a member of the Drift Stability Panel, for which the analyses were conducted.

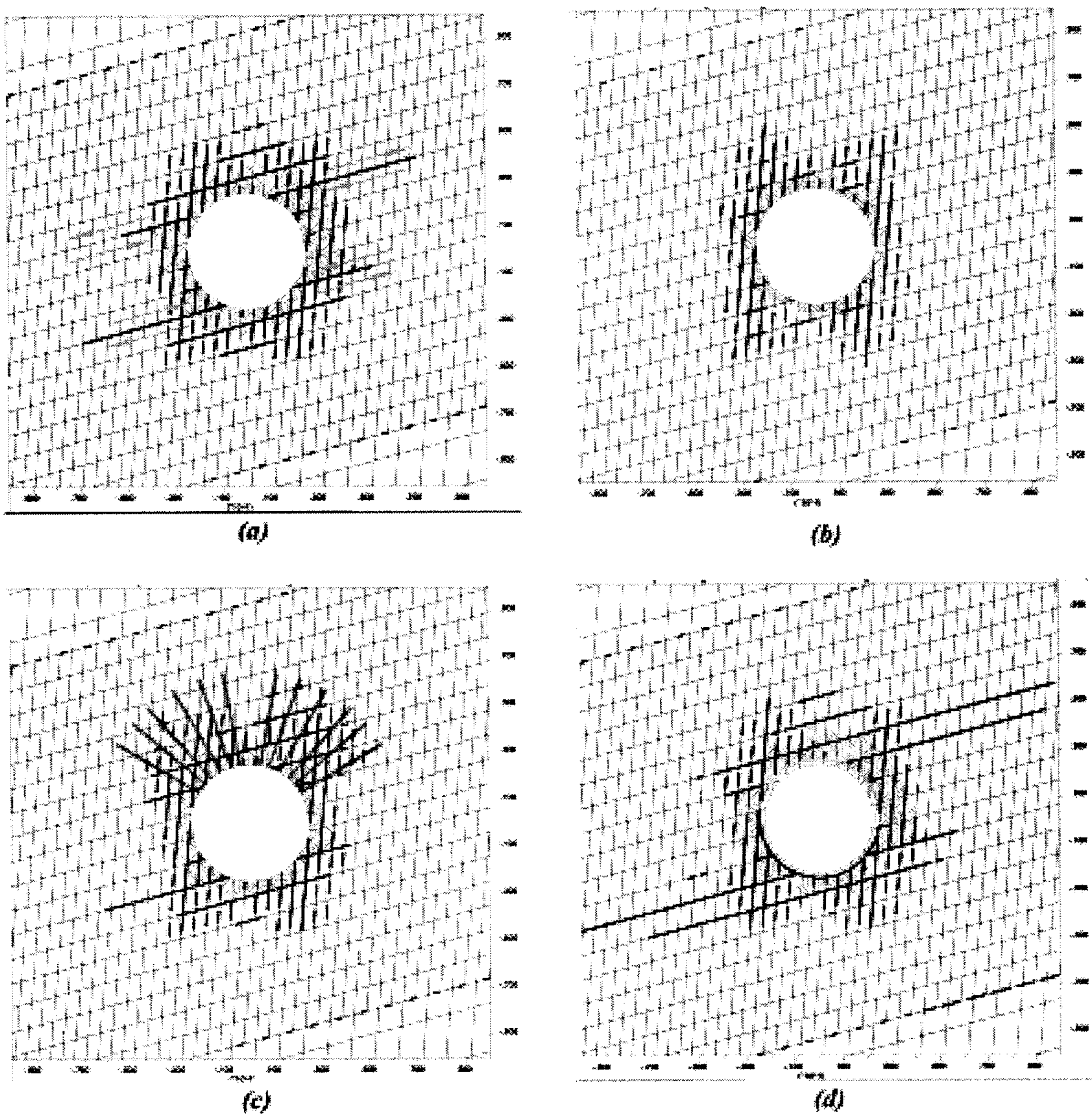
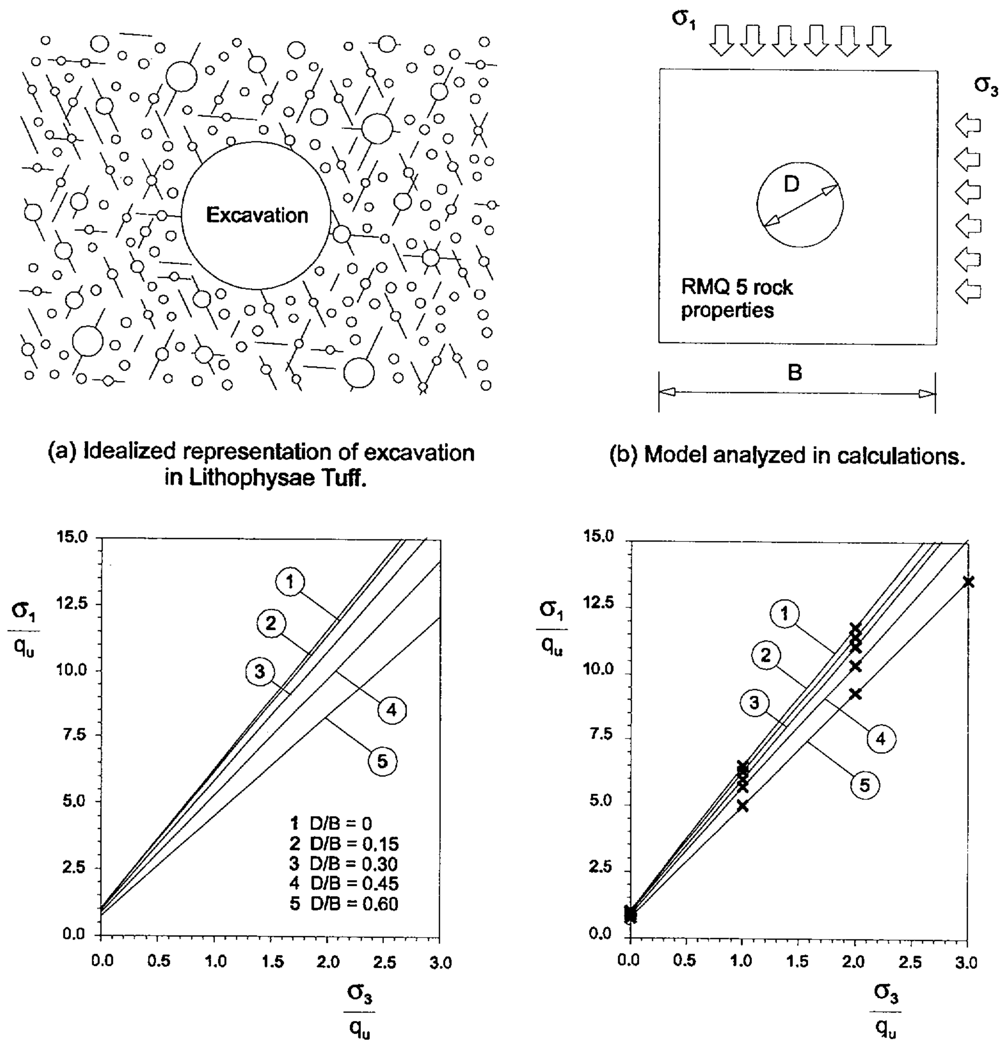


Figure 10. ***Effect of Long-term Degradation of Rock Joints Properties on Extension of Inelastic*** **Failed** ***Rock Zone for (a) Unsupported, Regularly Jointed Rock; (b) Unsupported, Non-persistent Jointing; (c) Reinforced by Jointed Rock Bolts; and (d) Supported by Elastic Concrete Support***



(a) Idealized representation of excavation in Lithophysae Tuff.

(b) Model analyzed in calculations.

(c) 'Tributary area' yield envelopes..

(d) FLAC^{3D} yield envelopes.

Note. Tributary area strength in (c) is calculated from the expressions,

$$\frac{\sigma_1}{q_u} = \left[1 - \frac{\pi}{4}\left(\frac{D}{B}\right)^2\right] K_p \frac{\sigma_3}{q_u} + \left[1 - \frac{\pi}{4}\left(\frac{D}{B}\right)^2\right] \qquad K_p = \frac{1 + \sin\phi}{1 - \sin\phi}$$

where q_u is the unconfined compressive strength and ϕ is the friction angle [with these intact rock properties, the yield envelope is given by line 1 in (c) and (d)].

The FLAC3D analysis also yielded the following results for the influence of the lithophysae on the overall modulus of deformation (E) of the lithophysae rock compared to the modulus of the rock without the lithopysae (E').

D/B	0.0	0.3	0.45	0.6
E/E'	1.0	0.96	0.89	0.78

Note E' in the FLAC3D analysis was 7.76GPa.

Figure 11. Predicted Rock Mass Strength (Mohr-Coulomb) Envelopes and Modulii of Deformation for Lithophysal Tuff (Intact rock between lithophysae is assumed to have RGM 5 mechanical properties.)

lithophysae and, eventually, the rock around the spherical periphery will begin to "collapse" into the lithophysal cavity.

The model analyzed is shown in Figure 11(b). B is assumed to be the width of a cubical region containing one cavity, diameter D. Various ratios of B:D were considered. The reduction in strength of the cube of rock containing the cavity, compared to the strength of a cube without a cavity (B/D = 0) is shown in Figures 11(c) and11 (d).

Two approaches are taken. In the so-called *tributary area* method (frequently used for room and pillar design in mines), it is assumed simply that the strength is reduced in proportion to the reduction in cross-sectional area of the center section of the cube containing the spherical cavity. In the second approach, a three-dimensional numerical analysis (FLAC3D) was carried out. The strength limit was assumed to be reached when inelastic deformation started at the wall of the sphere. Results are shown in Figures 11(c) and 11(d). Although the FLAC3D results indicate slightly higher strengths for a given cavity size, the difference between the two approaches is small (maximum about 18% for D/B = 0.6), and the tributary area approach is conservative (i.e., it underestimates the strength of the rock). Thus, it seems sufficient to use the tributary area method in calculations involving the rock-mass strength of the lithophysal zone.

The FLAC3D analysis also yielded results for the influence of the lithophysae on the overall modulus of deformation (E) of the lithophysal rock compared to the modulus of the rock without the lithophysae (E*). Results (tabulated in Figure 11) indicate that the reduction in E is also small, and follows a similar trend to that of the strength reduction.

It is recommended that laboratory tests be carried out on intact samples (taken between lithophysae) to establish the envelope corresponding to D/B = 0, and then to estimate an average value of D/B from exposures in drift walls. This information can then be used, with Figure 10(c), to establish an envelope for the rock mass strength.

Actual lithophysal voids tend to be ellipsoidal rather than perfectly spherical. Although it is feasible to generate ellipsoidal cavities and analyze them numerically, the effect of such cavities will depend on their distribution in size and orientation with respect to each other and to the applied stress field. As a first approach, over-conservative but simple approximation, the voids could be assumed to be "replaced" by spheres of diameters equal to the major axis of the ellipsoid. (A less conservative option would be to assume spheres of diameter equal to the mean of the major and minor axes of the ellipsoids.) The approximate expressions presented in Figure 11 could then be used.

Use of Concrete for Excavation Support

Concern has been expressed that the use of concrete, as is popular, in concrete and "shotcrete" linings and in the cement grouting of rock bolts[9] would result in a high pH of water entering the drift. This could have numerous adverse consequences (for example, on the radionuclide retardation capability of materials that may be placed below the waste packages, or

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[9] Note that resin grouts are not favored, as they are organic compounds.

that exist below the repository, e.g., zeolites) in order to retard the movement of radionuclides e.g., neptunium.

Discussions with concrete technologists reveal that it is possible to avoid high-pH water (e.g., by carbonating the cement, using carbon dioxide). The carbonation reaction has been studied extensively (it occurs naturally in concrete due to the effects of carbon dioxide in the atmosphere), and it appears possible to engineer a solution to avoid high-pH water. Also, the strength (and ductility) of concrete can be increased considerably compared to standard concretes traditionally used in construction. Although care should be taken to ensure that adverse effects are avoided, it is recommended that drift support designers not be prevented from taking advantage of the merits of shotcrete and grouted bolts, both of which could play a valuable role in drift support at Yucca Mountain.

Most of the designs showing precast concrete lining or steel sets in the (circular) drifts (admittedly idealized) indicate that the linings/sets are in intimate uniform contact with the drift wall. In reality, of course, there will be irregularities in the wall profile. Normally, these would be filled with cement grout to ensure that the lining is uniformly loaded. Sand *backpacking* can be substituted, but it is important that analysis of the lining support include consideration of the influence of such irregularities and fill methods on the bending stresses generated in the support during the thermal cycle.

The writer believes that a well-designed system of grouted rock bolts, mesh, and shotcrete will be sufficient to ensure stable openings during the preclosure period. Precast concrete linings or steel set supports, which would be very expensive, will not be needed.

Upper-Bound to Collapse Region

A simple estimate of the maximum extent of collapse around an unsupported tunnel can be made as follows.

Consider a circular tunnel, of radius a, surrounded by a circular zone of damaged rock, radius V. When rock is damaged, slip along joints and dilation occur, rock may collapse into the tunnel, etc. (i.e., the damaged rock will occupy a greater volume than when it was intact and undisturbed; it is said to undergo "bulking"). Let us assume that the rock is damaged to a radius b $(b > a)$. If we assume that the broken rock has a bulking factor (i.e., unit volume of unbroken rock occupies a volume $(1+k)$ in the broken state), we may determine the volume of unbroken rock in the annulus $(b - a)$ that, upon breaking, will fill the excavation. Thus, we have

$$\pi\,(b^2 - a^2)(1 + k) = \pi\, b^2$$

from which we obtain

$$\frac{b}{a} = \left(\frac{1+k}{k}\right)^{0.5}$$

For a value of $k = 10\%$ (10% to 25% is considered to cover most mining collapse situations), we find $b/a = 3.3$. For $k = 25\%$, $b = 2.2$.

Thus, the maximum possible extent of the damage zone around a repository tunnel will be of the order of three tunnel radii. Beyond this region, the rock will contain joints and fractures similar to those in the virgin rock mass. Hence, for calculation of post-thermal cycle water influx to the tunnel, such a model should suffice.

Heated drift experiments and niche tests are unlikely to resolve several important post-thermal cycle inflow issues. The effect of the thermal cycle on the mechanical properties of the rock mass, information that would have been very useful in drift stability analysis, appears to be a secondary consideration in these experiments compared to the hydrological issues. There has been no modeling of the effect of discrete jointing on rock mass behavior, for example. (Appendix II shows a preliminary study to illustrate what is possible.) Acoustic emission (micro-seismic) studies have only recently been added, and an opportunity to observe the rock-mass response from the onset of loading has been missed. Some microseismic equipment has now been installed, and data are being collected. Collection of such data can be very valuable in establishing which joints are slipping, and this information can be used to calibrate numerical models that contain such discrete features. (Figure 12 illustrates the microseismic network set up for the mine-by experiment at the Underground Research Laboratory in Canada, together with the locations of the microfracturing (detected by acoustic emission) induced by excavation. The network was installed before the mine-by excavation was started.)

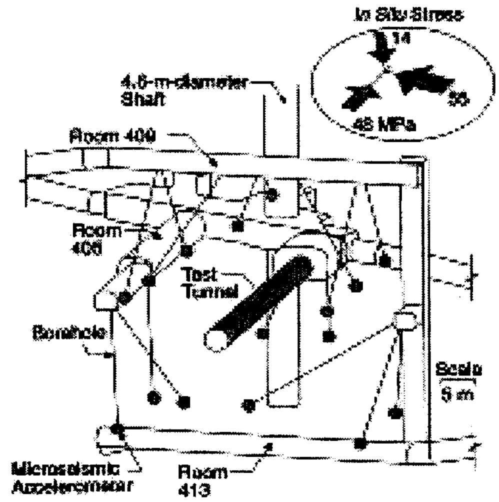

Figure 1: Layout of the 420 Level showing the location of the Mine-by test tunnel, the microseismic monitoring system and Room 405, the location of the borehole breakout study.

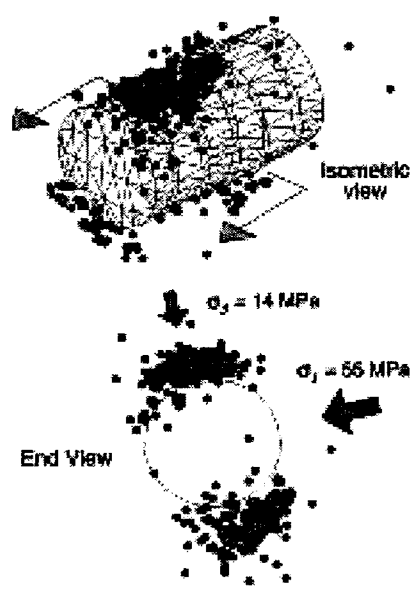

Figure 13: Location of microseismic events recorded after the excavation of a 1-m-long round in the test tunnel.

Figure 12. Mine-by Experiment, Underground Research Laboratory, Pinawa (Read and Martin, 1996)

References

Carter, Luther J., and Thomas H. Pigford 1998. "Getting YUCCA MOUNTAIN RIGHT," *Bull. Atomic Scientists*, 56—61 (March—April).

Electric Power Research Institute (EPRI) 1996. ***Analysis and Confirmation of the Robust Performance of the Barrier System within the Yucca Mountain Site***. Flow Division, Report TR107189 (3294-17), December.

Goodman, R. E. 1989. ***Introduction to Rock Mechanics***, 2nd ed. New York: Wiley.

Philip, J. R. 1990. "The Watertightness to Unsaturated Seepage of Underground Cavities and Tunnels," in ***GEOVAL-90 (Proceedings of the Symposium on Validation of Geosphere Flow and Transport Models, Stockholm, May 1990)***, pp. 301—308. Stockholm: SKI.

Philip, J. R., J. H. Knight, and R. T. Waechter. 1989. "Unsaturated Seepage and Subterranean Holes: Conspectus, and Exclusion Problem for Circular Cylindrical Cavities," *Water Resources Research*, ***25***, 16—28.

Potyondy, D., and P. Cundall. 1999. "Modeling of Notch Formation in the URL Mine-by Tunnel: Phase IV — Enhancements to the PFC Model of Rock," ICG, Report to Underground Research Laboratory. City: Atomic Energy of Canada Limited, March.

Read, R. S., and C. D. Martin. 1996. ***Technical Summary of AECL's Mine-by Experiment, Phase 1: Excavation Response***, AECL-11311, COG95-171. City: Atomic Energy of Canada Limited, February.

Roseboom, E. H. 1983. " Disposal of High-Level Nuclear Waste Above the Water Table in Arid Regions," Circular 903 City: U.S. Geological Survey.

Shoesmith, D. W., and M. Kolar. 1998 "Waste Package Performance," in ***Alternative Approaches to Assessing the Performance and Suitability of Yucca Mountain for Spent-Fuel Disposal***, Ch. 5, pp. 5-1 — 5-67, Report TR108732, City: Electric Power Research Institute (EPRI), November.

Stellavato, N., and P. Montazer. 1996. ***Nye County NVirRP Office Annual Report 1995/96***.

U.S. National Academy of Sciences/National Research Council (NAS/NRC). 1957. "The Disposal of Radioactive Waste on Land," Publication No. 519. Washington, D.C.

U.S. National Academy of Sciences/National Research Council (NAS/NRC). 1995. ***Technical Bases for a Yucca Mountain Standard***, TYMS. Washington, D.C.

U.S. Nuclear Regulatory Commission, Disposal of High-Level Radioactive Wastes in a Proposed Geological Repository at Yucca Mountain, Nevada, Proposed Rule, Federal Register Notice, February 22, 1999.

U.S. Department of Energy (U.S. DOE). 1998. VA reference design (CWRMS/M&O Report Design Feature Evaluation #25, Repository Horizon Elevation, April 2, 1999).

Appendix I

Steady Vertical Unsaturated Infiltration Through an Array of Horizontal Drifts

Assumptions

Validity of Richard's equation with equivalent unsaturated properties. Isothermal medium.

$$\nabla \cdot (\, \mathbf{K} K_r(\psi) \nabla(\psi + z)\,) = 0$$

ψ : relative pressure head ($\psi < 0$ unsaturated zone, $\psi > 0$ saturated zone) [m]
z : vertical coordinate [m]
$\mathbf{K}$: hydraulic conductivity tensor [m/s]
$K_r(\psi)$: relative hydraulic conductivity ($K_r < 1$ unsaturated zone, $K_r = 1$ saturated zone) [-]

Parametric model for unsaturated conductivity

van Genuchten : $$K_r(\psi) = \frac{1}{(\, 1 + |\alpha\psi|^n \,)^{m/2}} \left(1 - \left(1 - \frac{1}{1 + |\alpha\psi|^n} \right)^m \right)^2 \quad , \quad m = 1 - \frac{1}{n}$$

Exponential : $$K_r(\psi) = e^{\alpha\psi}$$

Assumed material properties

Matrix porosity : 0.1
Matrix permeability : $4 \cdot 10^{-18}$ m^2
Fracture frequency : 4.5 1/m
Fracture aperture : 54 µm

Matrix hydraulic conductivity (isotropic) K_{min} : $4 \cdot 10^{-11}$ m/s
Fracture hydraulic conductivity (cubic law) K_{max} : $5.85 \cdot 10^{-7}$ m/s
Homogeneous saturated and residual moisture θ_s, θ_r : 0.1, 0.01
van Genuchten model parameters α, n : 4 1/m, 2
Exponential model parameter α : 10 1/m

2D vertical equivalent hydraulic conductivity tensor (assuming vertical fractures)

$$\mathbf{K} = \begin{bmatrix} \beta K_{max} & 0 \\ 0 & K_{max} \end{bmatrix}$$

Anisotropy ratio β varied from 1 to K_{min}/K_{max}

Geometry

Drifts; diameter - 5 m ;spacing - 80 m (horizontal), 30 m (vertical)
Potential capture zone (per unit width) for a column of drifts : 80 m^2

Boundary conditions

Unsaturating infiltration rate at the surface (i): 50 mm/y ($1.37 \cdot 10^{-4}$ m/d , $1.59 \cdot 10^{-9}$ m/s)
Atmospheric seepage at the drifts
Static watertable level at – 200 m

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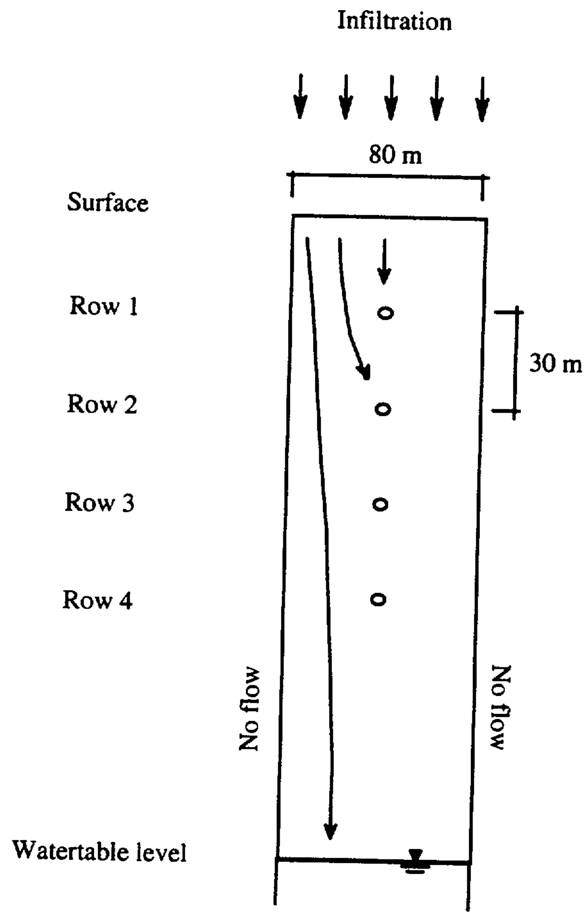
* i.e. 1096 x 10^{-5} m^3/d over the 80 m^2 potential capture area.

Sketch of the flow towards drifts in columns (with symetry conditions)



Discharge rates under steady state unsaturated conditions [10^{-5} m^3/d]

	Case 0 : $\beta = 1$ (isotropic)	Case 1 : $\beta = 10^{-1}$	Case 2 : $\beta = 10^{-2}$
Top infiltration	1096.0	1096.0	1096.0
Drifts at – 30 m	– 0.0	– 0.0	– 35.6
Drifts at – 60 m	– 0.0	– 0.0	– 0.0
Drifts at – 90 m	– 0.0	– 0.0	– 0.0
Drifts at – 120 m	– 0.0	– 0.0	– 0.0
Bottom drainage	– 1096.0	– 1096.0	– 1060.4

	Case 3 : $\beta = 10^{-3}$	Case 4 : $\beta = 10^{-4} (\approx K_{min}/K_{max})$	Case 5 : $\beta = 0$
Top infiltration	1096.0	1096.0	1096.0
Drifts at – 30 m	– 58.0	– 60.3	– 68.5
Drifts at – 60 m	– 6.1	– 7.0	– 0.0

Drifts at – 90 m	– 3.2	– 2.4	– 0.0
Drifts at – 120 m	– 2.9	– 2.0	– 0.0
Bottom drainage	– 1025.8	– 1024.3	– 1027.5

Remarks

Under unsaturating vertical infiltration, buried cavities may behave as obstacles to the flow and so increase water relative pressure head at parts of the cavity surface. Gravity dripping into the cavity occurs only at those points where the pressure head reaches the pressure inside the cavity (e.g., atmospheric pressure). Under uniform infiltration the first point reaching this pressure is the highest point of the cavity roof.

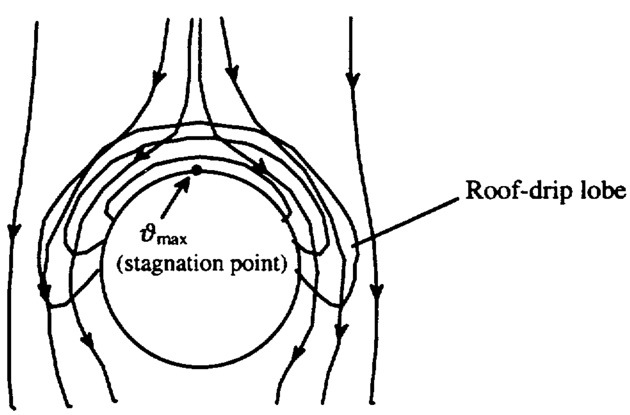
An analytical solution exists for horizontal cylindrical cavities in an <u>isotropic</u> medium (Philip and Knight, 1989, *Water Resources Research*, **25**, 16-28). Assuming an infinite vertical medium submitted to constant, uniform unsaturated seepage, the question as to whether or not water drips into a circular section centered at the origin is answered in a straightforward manner with the following simple rule

$$\text{if} \quad i < \frac{K_{sat}}{\vartheta_{max}(s)} \qquad \text{no dripping in the cavity, dripping otherwise}$$

In this (exact) formula i [m/s] is the specified uniform infiltration rate, K_{sat} [m/s] is the saturated isotropic hydraulic conductivity and ϑ_{max} is a normalized Kirchoff potential. Its value is maximum at the top of the circular section and can be approximated with excellent practical accuracy by

$$\vartheta_{max}(s) \approx \left| \begin{array}{ll} 2s + 1 & \text{, for small values of } s \\ 2(s + 1) & \text{, for large values of } s \end{array} \right. \qquad , \qquad s = \frac{\alpha D}{2}$$

where s is a dimensionless quantity defined by the decay parameter (α) of the Exponential model for the relative conductivity, and by the cavity diameter (D). Small s indicate capillarity dominated seepage, tending to divert water around the cavity, whereas gravity is dominant for larger values. Moreover, the larger the cavity the more vulnerable it is to water entry.



Iso-ϑ around the cavity and seepage flow lines

In the present situation (s = 10·5/2) no dripping occurs into the cavity since the above inequality is satisfied ($1.59\cdot10^{-9} < 5.58\cdot10^{-9}/52$). The infiltration rate could actually be increased by, roughly, a factor 10 before droplets form at the top of the cavity. Alternatively, the isotropic saturated hydraulic conductivity could be reduced, or the cavity diameter increased by the same factor, to produce dripping into the cavity.

These theoretical considerations explain why the drifts remain dry in Case 0 and to a certain extend in cases with mild anisotropy (i.e., Case 1). As anisotropy becomes larger, horizontal capillary flow becomes less significant and water cannot be diverted around the cavity surface with the same magnitude any more.

As a result, saturation increases and dripping starts in the first drift (Case 2), while the drifts below remain dry because the roof-drip lobes coming from above are too diffuse (capillarity is still active) to generate saturation conditions there.

At larger anisotropy ratios (Case 3 and Case 4) the lower drifts become gradually active, but in a manner that is not straightforward to understand. There are obviously highly non-linear effects (the decay coefficient α is rather large) combined to the anisotropy ratio. Numerical effects due, for instance, to mesh orientation and refinement around the drifts may also be present. However, several grid size were enforced (the finer with node spacing of the order of 0.2 m around the drifts) yielding the same type of results. More investigations (including analytical ones) are needed to understand the flow processes (e.g. use of finer meshes and various solution schemes, columns with more drifts, etc), particularly at high anisotropy ratios.

With zero horizontal conductivity (Case 5) the first drift theoretically captures the quantity of water given by *iD* (i.e., $68.5 \cdot 10^{-5}$ m^3/d in the present case) and by-passes the drifts vertically below.

Appendix II

Numerical Simulation of the Effects of Heating on the Permeability of a Jointed Rock Mass

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To be presented at the 9th ISRM Conference (Paris, August 1999)

Numerical simulation of the effects of heating on the permeability of a jointed rock mass

Simulation numérique des effets d'une augmentation de température sur la perméabilité d'une masse rocheuse fissurée

Numerische Simulation der Hitzeeinwirkung auf geklüftetes Gebirge

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ABSTRACT: One of the objectives of the Drift Scale Test (DST), currently underway at Yucca Mountain, USA, is to assess the effect of large-scale heating on the permeability of the rock mass. The DST is simulated using continuum and discontinuum models to predict the change in permeability in the rock mass surrounding the heated drift. The simulations show that heating will cause both reduction in permeability (in regions of increasing mean stress) and increase in permeability (in regions of non-linear shear deformation—slip). Although the elasto-plastic (ubiquitous joint) continuum model and the distinct element model (DEM) indicate similar regions of joint slip in the rock mass, the resulting change in permeability can be calculated much more easily from the DEM.

RÉSUMÉ: Un des objectifs de l'essai DST (Drift Scale Test), en cours au site de Yucca Mountain, Etats Unis, est l'évaluation de l'effet d'une variation thermique sur la perméabilité de la masse rocheuse, à l'échelle de la galerie. L'essai DST est simulé numériquement à l'aide de modèles continu et discontinu afin de prédire le changement de perméabilité de la masse rocheuse entourant la galerie lorsqu'elle est soumise à une augmentation de température. Les simulations numériques montrent que l'échauffement cause à la fois une réduction (dans les régions d'augmentation de la contrainte moyenne) et une augmentation de perméabilité (dans les régions de déformation non-linéaire en scisaillement–glissement). Bien que les modèles continu élastoplastique (ubiquitous joint) d'une part et d'élements distincts (DEM) d'autre part prédisent des zones similaires de glissement de joint dans la masse rocheuse, la méthode DEM se prête plus aisément au calcul des changements de perméabilité.

ZUSAMMENFASSUNG: Eine der Aufgabenstellungen des "Drift Scale Tests - DST", der gegenwärtig im Yucca Mountain Projekt in den USA durchgeführt wird, ist es, den Effekt von großräumiger Erhitzung auf die Permeabilität des Gebirges zu untersuchen. Der DST wurde durch Kontinuums- und Diskontinuumsmodelle simuliert, um die Änderungen der Permeabilität im Gebirge um den erhitzten Teil zu prognostizieren. Die Simulationen zeigen, daß die Erhitzung sowohl eine Reduzierung der Permeabilität (in Regionen erhöhter mittlerer Spannungen) als auch eine Erhöhung der Permeabilität (in Regionen nicht-linearer Scherdeformationen - "slip") bewirkt. Obwohl das elasto-plastische (verschmierte Klüfte) kontinuumsmechanische Modell und das Distinkt-Element-Modell (DEM) ähnliche Bereiche von Scherbewegungen auf Klüften ausweisen, kann die resultierende Änderung der Permeabilität über die DEM wesentlich einfacher bestimmt werden.

1 INTRODUCTION

A main objective of the ongoing Drift Scale Test (DST) at Yucca Mountain, Nevada, USA, is to assess the effect of large-scale heating (intended to simulate the heating produced by stored high level nuclear waste) on the permeability of the rock mass. The DST is conducted in fractured, densely welded, ash-flow tuff at the proposed repository horizon in Yucca Mountain. The permeability of this rock mass is controlled primarily by natural fractures in the rock; the matrix permeability is very small.

This paper discusses the results of numerical analyses carried out to examine the effect of heating around the DST on the change of permeability in the surrounding rock. Continuum models of a fractured medium (e.g. the ubiquitous joint model) provide reasonable approximation of the rock mass when: (1) the joint spacing is small relative to the characteristic dimensions of the problem, and (2) the joint properties are uniform (i.e. there are no joints in the set that have an aperture and transmissivity substantially greater than that of other joints). Determination of the constitutive relations needed to allow accurate prediction of the change in permeability of such a rock mass when deformed is especially difficult with continuum models. The relationship between deformation and permeability can be represented much more directly in models (such as the distinct element method), that simulate explicitly the effect of joints on deformation and fluid transport.

Given the actual geometry of the excavations and joints, rigorous interpretation of the effect of heating on joint aperture and permeability changes and flow in the drift experiment requires a three dimensional model. *3DEC* (Itasca Consulting Group, Inc. 1998a) was used to consider this influence. However, since a coupled thermo-mechanical-hydrological analysis of a fractured rock mass is computationally intensive, the main part of the analysis in this study has been carried out using the two-dimensional Universal Distinct Element Code, *UDEC* (Itasca Consulting Group, Inc. 1996). The continuum code *FLAC* (Itasca Consulting Group, Inc. 1998b) was also used to estimate the regions of non-linear deformation (i.e. the regions where the rock permeability changes) induced in the rock mass by heating. Comparison of results obtained using different models and codes (continuum; discontinuum, two-dimensional; three-dimensional) has proven to be very valuable in verifying the assumptions used in development of the analyses and may guide the use of particular models in further analysis.

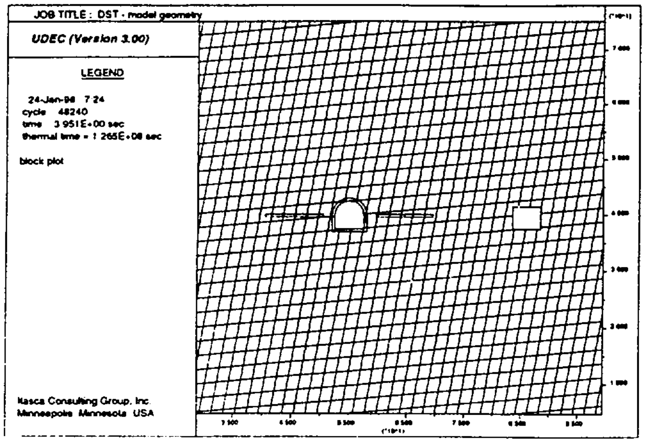

Figure 1. Geometry of the two-dimensional discontinuum model

2 DESCRIPTION OF THE MODELS

The heated drift is a 5 m × 5 m excavation of "horse-shoe" cross-section (see Fig. 1). The observation drift is rectangular, 5 m × 4 m in cross-section. A three-dimensional model of the DST was generated using *3DEC*. Figure 2 shows the lower half of this model (i.e. from the drift horizon downward). Three joint sets are represented. Joint set 1 has a dip of 77° and dip direction of 40°; set 2 has a dip of 80° and dip direction of 130°; set 3 has a dip of 25° and dip direction of 300° (Wagner 1996a). The joint spacing in each set is 10 m. The vertical cross-section, perpendicular to the axes of the drifts (from the *3DEC* model), coincides with the plane of the two-dimensional models used for simulation of the DST.

Figure 1 shows the joint sets 1 and 3 in the two-dimensional *UDEC* model. The joints in the two-dimensional model are spaced 2 m apart—i.e. much closer than the 10-m spacing in the three-dimensional model. (The coarser spacing in the *3DEC* model is dictated by the heavy computational demands of three-dimensional analysis.)

The rock was considered to be linearly elastic and isotropic, and to have the properties (Birkholzer & Tsang 1996) shown in Table 1. The response of the joints to deformation normal to the joint plane is assumed to be linearly elastic for compressive stresses (Joints can not sustain tension.); the response to shear deformation is assumed to be linearly elastic-perfectly plastic according to the Mohr-Coulomb slip condition. Slip of the joints is associated with dilation—i.e. joint opening.

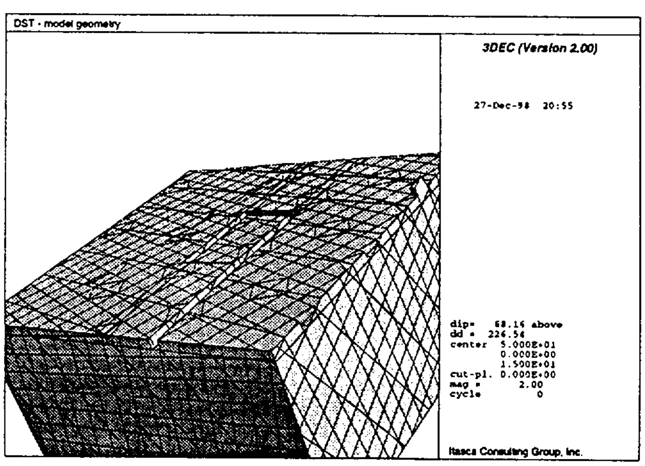

Figure 2. View of three-dimensional model (blocks above the drifts are hidden)

Table 1. Properties of the rock

Density, ρ	2540	kg/m^3
Young's modulus, E	32.4	GPa
Poisson's ratio, ν	0.17	
Thermal conductivity, k_t	1.67	W/m°K
Specific heat, c_t	928	J/kg°K
Coefficient of thermal expansion, α	10^{-5}	

"Fast paths", joints or fracture zones with much higher initial permeability (i.e. initial hydraulic aperture) than the other joints, are known to occur at Yucca Mountain. It is also expected that fast paths will be more compliant and weaker than the other joints. Two cases were considered in the discontinuum models: (1) all joints have the same properties, and (2) "fast-paths" are assumed to exist at several different locations relative to the heated drift. The properties of "typical" rock joints (Olsson & Brown 1997) used in the analysis are shown in Table 2.

Table 2. Properties of the rock joints

Normal stiffness, k_n	200	MPa/mm
Shear stiffness, k_s	150	MPa/mm
Cohesion, c	0.23	MPa
Friction angle, ϕ	42°	

For this analysis, the mechanical properties of the fast paths (shown in Tab. 3) are simulated by reducing the properties of "typical" joints—as can be seen by a comparison of Tables 3 and 2.

Table 3. Properties of fast paths

Normal stiffness, k_n	50	MPa/mm
Shear stiffness, k_s	50	MPa/mm
Cohesion, c	0.05	MPa
Friction angle, ϕ	25°	

The initial state of stress in the rock mass was assumed to be $\sigma_h = -5$ MPa, $\sigma_v = -10$ MPa at the drift level. The initial stresses vary as a function of elevation due to gravity, with a constant ratio maintained between the horizontal and vertical normal stresses. The initial temperature in the rock mass was taken to be constant, at 25°C throughout the model.

Thermal analysis of conductive heat transport was carried out for 4 years. An 800-W/m heat source, provided by heaters located in the square block at the floor of the heated drift, was simulated as a heat flux uniformly distributed along the boundary of the heated drift. The wing heaters are located symmetrically relative to the axis of the heated drift: a planar source of 125 W/m^2 is distributed between 4 m and 9 m from the drift axis, and a planar source of 175 W/m^2 is distributed between 9 m and 14 m distance from the drift axis (Wagner 1996b).

3 JOINT DILATANCY

Joint (normal and shear) stiffness and strength (cohesion and friction angle) are properties that affect the dependency of the permeability (of the joints and rock mass) on the imposed mechanical loading. However, the joint dilation angle ψ has the most profound effect on the dependence of the permeability to shear deformation of a rock joint.

The joint dilation angle, the measure of joint opening as a result of joint slip, is a function of:

1. shear deformation (Dilation is usually large during the initial slip deformation, decreasing with slip accumulation.); and
2. stress normal to the joint plane (confinement). (Dilation is a consequence of joint roughness. The relative movement of rock blocks cannot be strictly parallel to the plane of the joint between them, since joint roughness enforces some displacement normal to the joint plane. At very high normal stresses, the joint asperities can be sheared-off, resulting in a reduced or zero dilation angle.)

Olsson & Brown (1997) reported joint dilation angles measured on samples taken from the TSw2 geological unit at Yucca Mountain for different confinements. (TSw2 is the repository unit.) The measured dilation angles show large dispersion, varying between 1.11° and 33.4°. As a result, the relationship between confinement and dilation angle is unclear. Therefore, the first-order analyses were conducted using an upper value, $\psi = 30°$, and an average value, $\psi = 14°$, for the dilation angle. It was further assumed in these analyses that the dilation angle was constant, independent of the shear deformation or normal stress. The dilation angle for the fast paths was assumed to be equal to the dilation angle of "typical" joints.

3.1 Numerical Experiment

In order to establish a clearer understanding of the dependence of dilation to shear deformation and confinement for the range of values expected to occur in the model, numerical experiments were conducted to simulate shearing of a rough joint using a shear box—in a manner similar to that described by Cundall (1999). The results from the numerical experiments (i.e. the relationship between peak dilation angle, joint shear displacement, and normal stress) for TSw2 rock and joint conditions were then used in the *UDEC* simulation of the DST.

The micro-mechanical model of the shear box experiment using the Particle Flow Code—PFC^{2D} (Itasca Consulting Group, Inc. 1999), is shown in Figure 3. The bonded assembly of particles (Particles are bonded at contact points.) can be envisioned as a synthetic rock. By adjusting the contact stiffness (shear and normal) and strength (shear and tensile), this "rock" was made mechanically similar to the TSw2 rock. The length of the specimen in Figure 3 is 0.10 m, and the height is 0.04 m. The joint trace is indicated by the continuous black lines transecting the specimen from left to right. The particles at or adjacent to this line are left unbonded. The black particles along the boundary of the specimen are designated as the shear box. The shear box particles below the joint trace are fixed, while those above the trace are assigned a constant horizontal velocity. The joint trace was produced using the following decreasing power law power spectrum (Brown 1995):

$$G(k) = Ck^{\alpha} \tag{1}$$

where C is a constant; $k = 2\pi/\lambda$;λ is the wavelength; $\alpha = 5 + 2D$; and D is the fractal dimension of joint surface. Joint topography data provided by Olsson & Brown (1997) for specimen YM30 taken from the repository unit TSw2 were used. Numerical

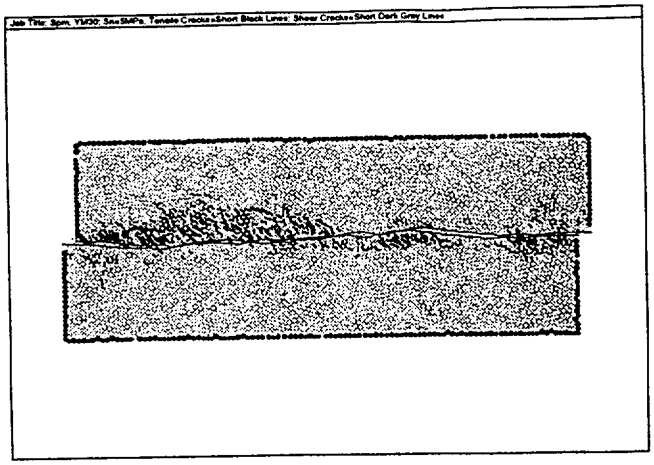

Figure 3. PFC model of a shear box test

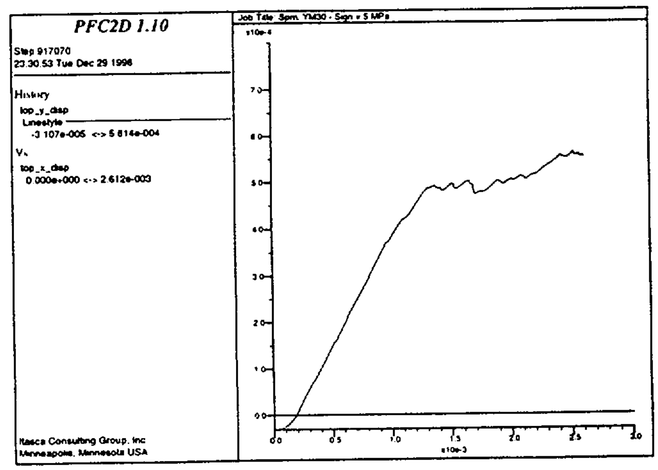

Figure 4. Vertical (m) versus shear displacements (m)

tests were conducted for normal stresses of 2.5, 5, 10, 15, 20, and 25 MPa. Figure 3 shows the specimen after a significant amount of shear for a normal stress of 5 MPa. The short black lines indicate locations of tensile cracks in the specimen, while the dark gray lines indicate shear cracks. Note that a significant amount of damage can be attributed to tensile cracking (i.e. particle contacts failing in tension). The test in Figure 3 predicted a peak shear strength of 6 MPa after 0.2 mm of shear displacement. Figure 4 shows normal displacement (m) (i.e. dilation) versus shear displacement (m) for the test in Figure 3. (Figure 4 suggests a peak dilation angle of 28°.)

The results from the numerical experiments were simplified as a bi-linear relationship between dilation and joint shear displacement. (This relationship is defined by a constant dilation angle and a shear displacement at which dilation becomes zero.) The dependence of the dilation angle and the zero dilation shear displacement on the confinement, as obtained from the numerical experiments (shown in Table 4), was implemented in the *UDEC* model of DST to provide a better approximation of the dilation behavior of the joints.

Table 4. Approximate relationship between joint dilation, zero dilation shear displacement, and normal stress for YM30

Normal stress (MPa)	2.5	5.0	2.5	10.0	15.0	25.0
Dilation angle (°)	42	28	16	15.5	13.0	12.0
Displacement (mm)	1.0	1.5	2.5	2.5	2.5	2.5

4 MODELING RESULTS

4.1 *Temperature fields*

It was assumed in all simulations (*FLAC*, *UDEC* and *3DEC*) that conduction is the only mode of heat transfer in the rock mass. In fact, boiling of pore water is likely to occur in the rock around the heated drift because of the high temperatures. This effect has been analyzed in models of heat and fluid transport by Buscheck (1998).

The temperature distributions due to heat conduction are almost identical for the continuum and discontinuum models. The contours (°C) after 4 years of heating are shown in Figure 5.

4.2 *Deformation in the two-dimensional continuum models*

The ubiquitous joint model is a continuum, elasto-plastic model in which an anisotropic strength of the rock mass is taken into account—i.e. there are predefined planes of weakness. The strength in the planes of weakness was assumed to be equal to the joint strength as given in Table 2. The markers shown in Figure 6 indicate slipping along the planes of weakness corresponding to sub-vertical joint set from Figure 1.

The ubiquitous joint model predicts the deformation and the region of joint slip in the rock mass. To assess the change in permeability produced by this deformation and slip, it is necessary to establish a constitutive relation between deformation (volumetric and shear) and the change in permeability. In the case of the distinct element method, the joint deformation is calculated, and it is usually assumed that the change in the joint hydraulic aperture is equal to joint normal displacement (i.e. closing and opening).

4.3 *Deformation in the two-dimensional discontinuum models*

The discontinuous model of the rock mass in which the joint properties are taken to be uniform shows a complex response to the perturbation induced by heating (Fig. 7). In general, it is possible to identify two regions exhibiting significantly different responses. In the immediate vicinity of the drift, the joints tend to close as a consequence of an increase in the compressive stress normal to the joint planes. Both the maximum closure and the region over which the joints close increase with the duration of heating. Joints from both sets (sub-vertical and sub-horizontal) tend to close, but the sub-vertical joints close more. Above and below the region of joint closure, the sub-vertical joint set dilates (opens) as a result of shear slip. Both the extent of the region where joints are opening and the value of the maximum opening increase as a function of the duration of heating. The maximum opening is more than twice as large as the maximum closure. The effect of dilation angle is significant. The maximum opening in the model with a 30° constant dilation angle (1.5 mm) is two to three times larger than in the model with a 14° dilation angle (0.6 mm). The maximum opening in Figure 7, which shows results for variable dilation angle calculated from the *PFC* model, is 1.2 mm. The regions of slip along joint set 1, as calculated in *UDEC*, agree remarkably well with the regions of plastic deformation indicated by the *FLAC* ubiquitous joint model.

The actual position of possible fast paths relative to the heated drift is unknown. However, the effect of the fast path was assessed by performing a series of simulations for three different assumed locations of the fast paths:

Case 1. The fast path passes through the heated drift.

Case 2. The fast path is offset approximately 15 m to the left of the axis of the heated drift.

Case 3. The fast path is offset approximately 15 m to the right of the axis of the heated drift.

The analysis shows that the effect of the fast path in case 1 is insignificant. The effects of the fast paths in cases 2 and 3 are dramatic. The joint opening and closure for case 3, after four years of heating, is shown in Figure 8. The maximum joint opening caused by slip in cases 2 and 3 is about 6 mm, compared to 1.5 mm in the model with uniform joint properties.

4.4 *Deformation in the three-dimensional discontinuum model*

The results of the three-dimensional model show that the two-dimensional model is an acceptable approximation of the deformation in the middle of the heated drift. However, deformation of joint set 2, which is neglected in the two-dimensional model, becomes important in the region close to the drift ends, where the temperature field is also three-dimensional.

5 CONCLUSIONS

Comparison of the results of different computational models used to predict the thermo-mechanical response of a jointed rock mass

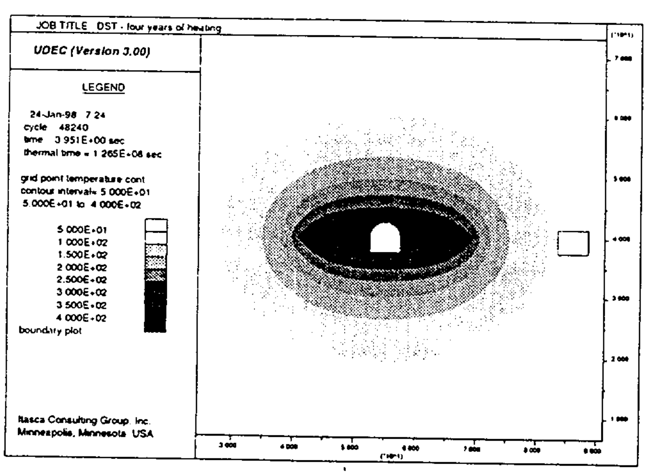

Figure 5. Temperature contours (°C) after 4 years of heating

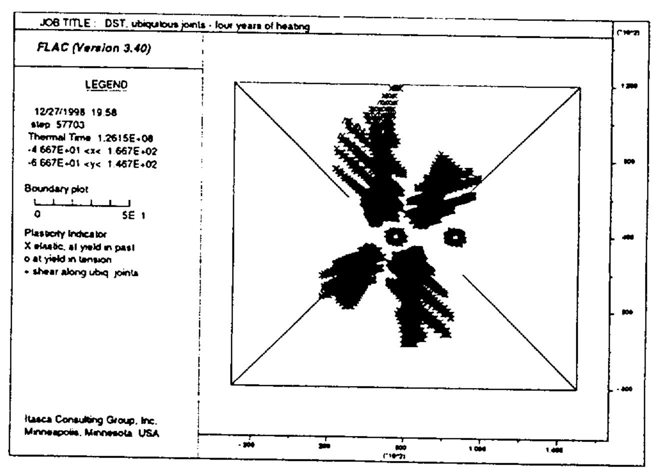

Figure 6. Indicators of slip in the ubiquitous joint model after 4 years of heating

in the vicinity of the DST, indicate the following.

1. Continuum, elasto-plastic ubiquitous joint models give a good prediction of the regions in the rock mass over which the joints slip. However, to calculate permeability change as a result of calculated deformation, a constitutive model that relates both volumetric and shear (elastic and plastic) strains to permeability change is required.
2. Discontinuum models are the most effective way to simulate the effects of heating (or any mechanical deformation) on change in permeability of a jointed rock mass. Constitutive relations are also required, but they are more straightforward than in the case of the equivalent continuum. Joint dilation angle and its dependence on accumulated slip and normal stress are important parameters that define the change in permeability produced by joint slip.
3. The two-dimensional model is an acceptable approximation of the deformation in the middle of the drift, even for the case in which orientation of the joints relative to the drifts' axes is slightly oblique.
4. Three-dimensional effects (particularly the deformation of the joint set neglected in the two-dimensional model) become important close to the end of the drift.

The various analyses described above have been used to illustrate the effects of large-scale heating on the hydrological conditions in the rock mass around the drifts in the DST. Increase in temperature produces different effects on the deformation of the rock joints (i.e. both closure and separation) in different regions of the rock mass. In general, shear stresses cause slip on the sub-vertical joints away from the drift, while increase in confinement causes closure of the joints (The sub-vertical joints close more.) in the vicinity of the heated drift. Both regions of opening and closure, and the maximum values of opening and closure in these regions are functions of several parameters, including: (1) intensity of thermal loading, and (2) properties of the rock mass and rock joints (e.g. stiffness, strength, dilation angle, orientation and spacing of joints). The effect of the deformation on the permeability of the rock mass is even stronger in the case when a fast path crosses the regions of large shear stresses induced by heating. The shear deformation and slip localize along the fast path. If a constant (independent of the magnitude of slip and the confinement)

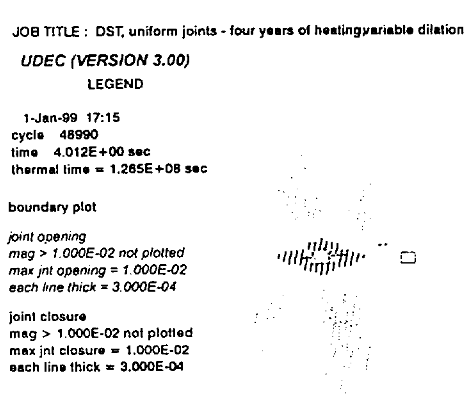

Figure 7. Uniform joint properties, variable dilation angle – opening and closure (m) of joints after 4 years of heating

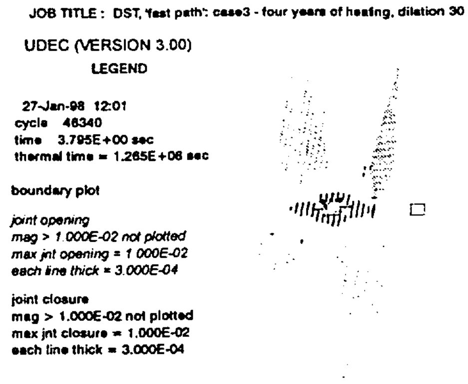

Figure 8. Fast path, case 3, dilation angle 30° – opening and closure (m) of joints after 4 years of heating

dilation angle of 30° is assumed, the opening of the fast path is of the order of six millimeters.

ACKNOWLEDGEMENT

The advice and interest of Professor P.A.Witherspoon, who suggested these studies, is gratefully acknowledged.

REFERENCES

Birkholzer, J. T. & Tsang, Y. W. 1996. Forecast of thermal-hydrological conditions and air injection test results of the single heater test. Technical report, Earth Sciences Division, Lawrence Berkeley National Laboratory.

Brown, S. R. 1995. Simple mathematical model of a rough fracture. *J. Geophys. Res. 100*, 5941–5952.

Buscheck, T. A. 1998. Private communications.

Cundall, P. A. 1999. Numerical experiments on rough joints in shear using a bonded particle model. In *Lecture Notes for Earth Sciences*. Springer. To be published.

Itasca Consulting Group, Inc. 1996. *UDEC (Universal Distinct Element Code) Version 3.0*. Minneapolis.

Itasca Consulting Group, Inc. 1998a. *3DEC (3-dimensional Distinct Element Code) Version 2.0*. Minneapolis.

Itasca Consulting Group, Inc. 1998b. *FLAC (Fast Lagrangian Analysis of Continua) Version 3.4*. Minneapolis.

Itasca Consulting Group, Inc. 1999. *PFC^{2D} Particle Flow Code in 2 Dimensions Version 2.0*. Minneapolis.

Olsson, W. A. & Brown, S. R. 1997. Mechanical properties of fractures from drillholes UE25-NRG-4, USW-NRG-6, USW-NRG-7, USW-SD-9 at Yucca Mountain, Nevada. Technical Report SAND95-1736, Geomechanics Department, Sandia National Laboratories.

Wagner, R. A. 1996a. Characterization of the ESF thermal test area. Technical Report B00000000-01717-5705-00047 REV01, TRW Environmental Safety Systems, Inc.

Wagner, R. A. 1996b. Test design, plans and layout. Report for the ESF thermal test. Technical Report BAB000000-01717-4600-00025 REV01 prepared for DOE, TRW Environmental Safety Systems, Inc.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

December 23, 1999

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: ADVISORY COMMITTEE ON NUCLEAR WASTE ROUND-TABLE DISCUSSION WITH YUCCA MOUNTAIN STAKEHOLDERS ON THE ROLE OF SAFETY ASSESSMENT IN REGULATORY DECISION-MAKING – OBSERVATIONS AND RECOMMENDATIONS

The Advisory Committee on Nuclear Waste (ACNW) hosted a round-table discussion on the role of safety assessment in regulatory decision-making on October 12, 1999, in Las Vegas, Nevada. Our objective in holding the round-table discussion was to enhance our own capability to communicate technical issues and to develop ideas about how to improve effective public participation in the NRC's regulatory process. We also hoped to strengthen our relationship with Nevada stakeholders and clarify our role as an independent technical oversight body to the NRC. The ACNW met with stakeholders again on the evening of October 12th to hear their concerns directly and to clarify the roles of the ACNW and the NRC. The NRC staff participated in both meetings.

In this letter we make some general observations about risk communication and risk perceptions, and convey three specific observations and recommendations. We also summarize input from stakeholders.

In support of the agency's goal to inspire public confidence in the regulatory process, the ACNW identified risk communication as one of its first-tier priority topics in its 1999 action plan. The ACNW believes that risk communication should be an essential element of risk-informed, performance-based regulation.

BACKGROUND

The NRC states in its strategic plan[1] that building and maintaining public trust is critical for carrying out its mission and achieving its vision, and that involving stakeholders in the conduct of NRC business is essential for inspiring public confidence in the agency's decisions and

[1]NRC Strategic Plan, September 1997.

actions. The NRC identifies an overarching management goal of inspiring public confidence by providing the public, those it regulates, and other stakeholders in the national and international community with clear and accurate information about, and a meaningful role in, its regulatory program.

The ACNW identifies in its 1999 action plan[2] a goal of supporting the NRC in improving public involvement. On the basis of what we learned from our international technical exchange meeting on nuclear waste held last year and from a meeting last year with Yucca Mountain stakeholders, the ACNW alerted the Commission that public involvement and public confidence may be the biggest impediments to progress in radioactive waste disposal worldwide.[3]

In exploring this important topic, the ACNW became familiar with ongoing public communication and public involvement initiatives internal and external to the NRC. The ACNW learned about new NRC communication and outreach efforts, as well as Environmental Protection Agency (EPA) and Nuclear Energy Institute (NEI) risk communication and public involvement initiatives and lessons learned. The ACNW also received training in risk communication and initiated the round-table discussion and public meeting with stakeholders in the Yucca Mountain area. These activities form the basis for our observations and recommendations, which follow.

GENERAL OBSERVATIONS

The stakeholders at the round-table meeting and in the audience contributed valuable information to the discussion. Participants in the round-table meeting were ACNW members and representatives from the State of Nevada's Agency for Nuclear Projects, the Environmental Protection Agency, Sandia National Laboratories, the Department of Energy, the Management and Operation (M&O) contractor for Yucca Mountain, Nevada's Clark, Eureka, Nye, and Lincoln counties, the Yucca Mountain Study Committee, the National Congress of American Indians, the Nevada Nuclear Waste Task Force, and some members of the general public.

The Committee offers the following general observations relating to the topic of the working group session.

1. Risk communication is the process of effectively exchanging information about risk with the public. It involves listening to the views and concerns of the public and stakeholders. It involves creating opportunities for the public to contribute to the regulatory decision-making process, not just to review documents.

2. The subject of risk has many facets, of which the principal ones are risk assessment, risk management, and risk communication. Risk communication, like assessment and management, should be an integral part of a risk-informed and performance-based regulatory process.

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[2]Advisory Committee on Nuclear Waste 1999 Action Plan and Priority Issues, January 22, 1999.

[3]ACNW Visit to German Waste Isolation Authorities and Facilities, September 14–18, 1999, General Observations and Impressions, dated January 27, 1999.

3. Stakeholder and public understanding of the NRC's use of risk assessment would be greatly enhanced if the process were transparent and made available to the public with appropriate opportunities for effective participation.

4. Unlike the public living near NRC-licensed nuclear sites, most Nevadans have little or no experience with the NRC and its way of regulating nuclear facilities. For this reason, the NRC needs to place major emphasis on informing the local public about the licensing of civilian nuclear facilities. The emphasis should not only be placed on the technical issues of safety, but on the comprehensive process the NRC employs to provide reasonable assurance about safety, including the use of confirmatory safety assessment.

STAKEHOLDER INPUT

Some of the perceptions of the different stakeholder and public groups are summarized below. Also below are comments pertaining to the subjects of risk communication and public involvement; the role and use of performance assessment; perceptions about the NRC's role, process, and regulations; the ongoing disagreement between the EPA and the NRC over the HLW standard; issues associated with transportation of HLW and the Yucca Mountain repository Draft Environmental Impact Statement (DEIS); and other subjects. Some representatives of the state and counties and members of the "public" perceived the following about the NRC:

- NRC's attempt at "risk communication" is disingenuous because of a lack of opportunities to influence NRC's options and decisions;
- The NRC relaxed regulatory requirements to ensure that the Yucca Mountain repository can be licensed (State);
- The NRC and the DOE have a strong camaraderie and a common language, and have a common interest in getting the repository licensed, that is, the NRC will not challenge the DOE (State, counties, the public);
- The disagreement between the EPA and the NRC undermines public understanding and trust in the NRC (State, counties, the public);
- The NRC has not justified its position against ground-water protection, and appears to want a less stringent standard than does the EPA because the proposed Yucca Mountain repository cannot meet the proposed EPA standards (State);
- The NRC does not have a clear "bottom-line" as to what it would take to reject the Yucca Mountain site (some members of the public);
- Some public groups are convinced that the NRC would never reject an application.
- Public participation has no impact on the NRC decision-making process and public input is not accepted (State, counties, the public);
- Once the NRC adopts the DEIS, it will not raise issues during its license application review or impose licensing conditions that are not contained in the scope of the DEIS (counties);
- Transportation is always an afterthought, and DOE will not consider the details of transportation until the repository is already decided upon (counties);
- The citizens of Nevada will not be given the same level of protection as what was given to the citizens of New Mexico in connection with the Waste Isolation Pilot Project (WIPP);

- A representative from the *Las Vegas Sun* asked how a citizen can participate in the risk assessment process itself, especially in identifying elements that should be analyzed; and
- The NRC needs to have a greater presence in Nevada.

SPECIFIC OBSERVATIONS AND RECOMMENDATIONS

Observation 1

Experience in other waste programs[4] and current research suggest that public confidence will likely depend more on the process of decision-making than on the scientific evidence used to support the conclusions. Pielke, et al.,[5] suggest that "the key to effective decision-making for any environmental problem lies in improving the decision environment itself, with the goal of making good decisions rather than good predictions.... In the absence of an integrated and open decision environment, the scientific merit of predictions can be rendered politically irrelevant...," and "In a healthy prediction process, stakeholders must question predictions, and predictions must be transparent, and assumptions and limitations forthrightly discussed."

Distrust in the Yucca Mountain performance assessment (PA) process was conveyed during the round-table discussions. The State representative noted that DOE has historically not accepted information or suggestions from people outside the DOE program concerning what to analyze in PAs. He believes that it is hard to trust the DOE's PA results when for years the DOE has acted confident about its understanding of the physical system at Yucca Mountain, but has been proved wrong time and again. A DOE representative indicated that the DOE has learned from conducting random focus groups around the country that people's greatest concern is whether the risk assessors have thought of everything that could go wrong. Yucca Mountain stakeholders also voiced this concern and the concern that in their desire to license a repository at Yucca Mountain, both the NRC and DOE may lack proper incentive to discover fatal flaws in the performance of the repository.

Recommendation 1

The NRC should evaluate the feasibility of directly involving the public in conducting its confirmatory performance assessment analyses for review of the DOE's total system performance assessment for the Site Recommendation and License Application. This would include the NRC's soliciting stakeholder ideas about what to consider in the analyses, and willingness to expose its total performance assessment analyses to the public for questioning. A similar process was used for conducting the PA for the WIPP site.[6] By including the public's concerns about what can go wrong in the NRC's independent analysis, the NRC could enhance its credibility and gain greater trust and confidence in its licensing process.

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[4]Presentation by Paul Davis on the WIPP program during the October 12, 1999, round-table discussion.

[5]Roger Pielke, D. Sarewitz, R. Byerly, Jr., and D. Jamieson, "Prediction in Earth Sciences and Environmental Policy Making," *EOS*, Volume 80, No. 28, July 13, 1999.

[6]Davis, op. cit.

Observation 2

Improving the practice of Risk-Informed, Performance-Based (RIPB) decision-making by ensuring that risk assessment results and risk management decisions are transparent is a critical step for facilitating stakeholder involvement and possibly gaining public confidence in the NRC's process. An RIPB framework should produce a more open, transparent, and consistent approach to risk-based decision-making, which, should foster greater visibility, access to, understanding of, and opportunity to participate in, the NRC's regulatory process. The process of becoming RIPB should help to create a healthy environment for decision-making that enables the public and other stakeholders to become involved more meaningfully in the regulatory process.

Recommendation 2

The NRC should focus on achieving greater consistency, clarity, and transparency in how it uses risk assessments across all of its waste programs in its decision-making process. For example, the NRC should clarify the extent to which it will rely solely on PA to make a regulatory decision about a Yucca Mountain repository and how and whether NRC will consider additional information.

Observation 3

One of the largest stakeholder concerns relates to transportation of HLW. Some members of the public fear that transportation accidents translate into nuclear disasters and to economic loss in affected communities. The absence of comprehensive and transparent risk assessments of nuclear waste transportation is a classic example of a failure in risk communication. It is not even clear whose role it is to evaluate and regulate the risk of the entire transportation system. Also of concern may be a real or apparent lack of integration in evaluating routing decisions about ongoing shipments of low-level waste (LLW) to the Nevada test site and proposed shipments of high-level waste (HLW) to Yucca Mountain.

Recommendation 3

The ACNW recommends that the NRC take the lead in clarifying the roles of the different agencies involved in the transportation of HLW, LLW, and in emergency response. The NRC should seek authority to require DOE to submit a comprehensive assessment of transportation risk at the time of the license application, so that this information can be considered as part of the overall licensing decision regarding Yucca Mountain.

Sincerely,

B. John Garrick
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

January 11, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: COMMENTS ON THE IMPORTANCE OF CHEMISTRY IN THE NEAR FIELD TO DOE's YUCCA MOUNTAIN REPOSITORY LICENSE APPLICATION

This letter provides comments and recommendations on the importance of chemical phenomena in the near field to the U.S. Department of Energy's (DOE's) Yucca Mountain Repository proposed License Application (LA). This letter also transmits the attached "white paper" by Raymond Wymer titled "Chemistry in the Proposed Yucca Mountain Repository Near Field." In his paper Dr. Wymer examines some chemical aspects of the near field of the Yucca Mountain repository with particular reference to chemical effects in engineered barriers, to the formation and dissolution of solids, and to corrosion. The potential contributions of intentionally added chemical materials to the near field for the control of radionuclide release and transport from the waste forms are discussed, along with the implications for possible NRC studies.

The recommendations are based on briefings by both NRC staff and DOE on DOE's Viability Assessment; on a working group meeting on the Near Field Environment and Performance of Engineered Barriers in the Yucca Mountain Repository held June 10–11, 1998, by Advisory Committee on Nuclear Waste (ACNW); and on discussions during a visit to the Center for Nuclear Waste Regulatory Analyses (CNWRA) June 28–30, 1999.

White Paper on Chemistry in the Proposed Yucca Mountain Repository Near Field

In the attached paper, Dr. Wymer discusses the potential advantages of using backfill in the drifts and in waste packages as a means of significantly retarding radionuclide transport by sorption and precipitation reactions. He reviews the status of chemical studies on the formation and dissolution of secondary phases at the waste form-water interface and discusses the potential importance of secondary phases in limiting release of radionuclides from the waste. The paper discusses the reliance by DOE on use of corrosion-resistant metals for drip shields, waste packages, and the zircaloy cladding of the spent fuel for preventing premature release of radionuclides from the repository.

DOE may use backfill or take credit for certain chemical processes in its analysis for its LA. Consequently, NRC should be prepared to analyze those aspects of an LA.

The problem of extrapolating corrosion data to the l0,000-year horizon makes it important to have as complete an understanding as possible of the important phenomena. For example, there is a temperature regime where corrosion of the metals of interest is much more severe than at temperatures anywhere outside that regime. NRC and CNWRA are at the forefront of corrosion studies related to the repository. This activity should continue.

Recommendations

Recommendation 1

The NRC staff should conduct scoping calculations of the importance of backfill to modify the chemical environment and to act as an attenuating agent for released radionuclides. If it is determined from these calculations that the use of backfill or the effects of corrosion products can have an important effect on performance, then more detailed analyses should be requested of the applicant.

Recommendation 2

The NRC staff should continue to work on the role of secondary phases in attenuating radionuclide releases. In particular, we recommend continued work on natural analogs, such as the Peña Blanca site.

Recommendation 3

The NRC staff should continue to collect as much confirmatory data as possible on the corrosion rates and mechanisms of corrosion of the drip shield, the waste packages, and the zircaloy cladding of the spent fuel under the range of expected conditions.

Recommendation 4

The NRC staff needs to continue and refine careful analysis of the role of coupled processes in repository performance as part of its development of review capabilities. Because of the complexity of the models and the abstraction of these models into the Total System Performance Assessment, the key focus is to ensure that an important coupled process is not left out of the performance assessment. The effects of temperature will be of particular importance.

ACNW will continue to follow developments in near-field chemistry of the proposed Yucca Mountain Repository.

Sincerely,

B. John Garrick
Chairman

Attachment:
R. G. Wymer, Member, Advisory Committee on Nuclear Waste and Andrew C. Campbell, Advisory Committee on Nuclear Waste, Senior Staff Scientist, "Chemistry in the Proposed Yucca Mountain Repository Near Field," dated January 11, 2000.

January 11, 2000

CHEMISTRY IN THE PROPOSED YUCCA MOUNTAIN REPOSITORY NEAR FIELD

R.G. Wymer, ACNW Member
Andrew C. Campbell, ACNW Senior Staff Scientist

Executive Summary

This paper focuses on select aspects of the near-field chemistry in the proposed Yucca Mountain high-level waste (HLW) repository. The goal is to identify areas where additional studies could help the Nuclear Regulatory Commission (NRC) position itself better to respond to changes in repository design. In this paper we focus on the portion of the near field within the confines of the drift walls. The near-field chemistry can be altered by chemical reactions that could occur by the interaction of percolating water with materials contained within the repository (e.g., backfill in the proposed) Enhanced Design Alternative II (EDA–II) design or concrete in drift liners that had been proposed as part of the Viability Assessment (VA) design. These chemical reactions can have an important effect on the potential release of radionuclides from the repository and ultimately its performance. A key area that the paper will consider is the possible use of chemically engineered backfills to alter near-field chemical properties to minimize the potential release of radionuclides and reduce long-term uncertainties in performance. Water composition, air composition and circulation, and temperature are discussed in the context of their effects on chemistry and on coupled effects in the repository near field. The importance of modeling to the license application and the influence of chemistry of the drift environment on modeling are also discussed.

The Role of Chemistry in Repository Licensing

The key issue from a regulatory perspective is the adequacy of information and models to make decisions about the safety of the repository. There will never be a complete knowledge base that covers all near-field chemistry issues. Because there are many deficiencies both in the availability of relevant information and in incorporating available information from the literature on chemistry, natural analogs, and experimental studies into databases, NRC will have to decide what information is essential for making regulatory decisions about the Site Recommendation and License Application. In particular, NRC will need to specify what information is critical for making a determination of reasonable assurance as it develops the Yucca Mountain Review Plan for the HLW repository.

The Near-Field Environment and Coupled Processes

Water is the *sine qua non* of chemical reactions in the near field, including interactions with engineered barriers, corrosion, solids dissolution, and radionuclide transport. There may be little that can be done in a practical way about controlling the long-term uncertainties in the amount of water that contacts the repository, although some possible solutions to this problem were examined in the white paper on Engineered Barriers at Yucca Mountain by Charles Fairhurst (1999). The chemical conditions of water reacting with materials in the near field may also be amenable to some control. For example, backfill, which is proposed in U. S. Department of Energy's most recent design, could be engineered to have certain chemical properties that can influence the near-field chemical environment in a positive way. From another perspective, engineering of the near-field environment should be done so as to avoid placing materials in the repository that could lead to increased uncertainties for radionuclide releases. An example is the use of large amounts of concrete as drift liners in the VA design.

Air, which is about 21 percent oxygen, plays a major role in promoting degradation processes that can release radionuclides from the repository. Because the proposed HLW repository at Yucca Mountain is in the unsaturated zone, the availability of significant air trapped in rock pores and fractures and the fluxes of air through fractures in the mountain (e.g., from barometric pumping) help to maintain an oxidizing environment.[1] This plays a dominant role in dissolving uranium dioxide in spent fuel and releasing fission products and actinide elements contained in the matrix. Radiolysis also may play a significant role in maintaining oxidizing conditions inside the waste package. In this environment, two of the important radionuclides (technetium and neptunium) exist in a more mobile state than if reducing conditions prevailed. In addition, the presence of carbon dioxide may lead to the formation of actinide carbonate complexes, especially uranium, that are also more amenable to transport with water. Various proposals for an open or closed repository after it has been filled with waste may also have a powerful influence on the potential for oxidation reactions in the repository near field.

Temperature in the repository, or more precisely the high temperature thermal pulse generated in the near-field environment by radioactive decay heat, is affected by the placement and areal extent of the waste packages in the repository. The thermal pulse can be greatly moderated by forced circulation of air at relatively high volumes during the open period of the repository, since a significant portion of the heat-producing decay reactions take place during the first few hundred years. Although the primary function of increased air circulation in the new DOE EDA–II design concept is to remove heat, some water will also be removed by evaporation. There is an important coupling of the effects of waterflow, airflow, and temperature with the near-field chemistry of the repository (i.e., coupled Thermal Hydrological Chemical [THC] processes); these effects are bound up inextricably with the repository design. Because reaction rates are controlled to a significant degree by temperature, a major goal of engineering to control the thermal loading of the repository is to reduce the uncertainties of coupled THC processes, which could result in changes in permeability in the near field and enhanced corrosion of the waste packages.

The Effects of Repository Materials on Near-Field Chemistry

Engineered barriers in the near field may be used to help control radionuclide release and transport and to limit degradation of waste packages, spent fuel cladding, and waste forms. Of particular chemical interest are drift backfill materials (and possibly materials introduced into the waste package) that could be used to buffer the near-field chemistry and constrain the formation of secondary phases so as to limit the release rate of radionuclides. There is a large range of potential backfill materials, each having advantages and disadvantages. The materials may include a variety of compounds that may act as reducing agents, precipitating agents, or scavengers for important radionuclides, such as technetium and neptunium. Backfill materials that have been proposed for nuclear waste repositories include crushed tuff, calcite, and quartzite sands (Yucca Mountain); various types of clays including bentonite and zeolites (saturated zone repositories proposed worldwide); and magnesium oxide (MgO) in the Waste Isolation Pilot Plant (WIPP). Depleted uranium has also been proposed as a backfill to help limit the dissolution of the uranium dioxide (UO_2) in the fuel and ensure that any mobilization of fissile material cannot form a critical mass (Forsberg,

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[1] The Yucca Mountain environment has a relatively high "redox potential." The redox potential of a system refers to its tendency to oxidize or reduce chemical species. Many important chemical processes in the repository will occur due to changes in redox potential. Uranium dioxide (UO_2) in spent fuel exists in the +4 valence state as it comes out of the reactor. Over time, as it is exposed to air and water containing oxygen and other oxidizing chemical species generated by radiolysis, the UO_2 becomes "oxidized" to the +6 valence state, which forms more soluble chemical compounds than the reduced form does. Hence oxidation of spent fuel results in the dissolution of the uranium matrix and the release of radiounuclides.

(UO_2) in the fuel and ensure that any mobilization of fissile material cannot form a critical mass (Forsberg, 1997). Addition of backfill to the waste package could put material in the immediate vicinity of the waste form and thus increase the efficacy of chemical actions; however, the thermal impact on cladding due to the insulating effect would have to be considered.

Formation and Dissolution of Secondary Solid Phases

The formation and dissolution of secondary radionuclide compounds that can form as spent fuel and other waste forms dissolve and reprecipitate as solid compounds are likely to be major determinants in the rate of radionuclide release from the proposed repository. In addition, radionuclide transport will be strongly affected by chemical species formed at the waste form-aqueous phase interface. Analyses made to date of both solids formation and radionuclide dissolution are limited because of the complexity of the chemical systems involved and the deficiencies of thermodynamic and kinetic databases used in the calculations for chemical reactions, solubilities, and new phase formation. Lack of specific chemical knowledge is sometimes treated in the modeling studies by making bounding-case assumptions. For example, the dissolution model that DOE used in the Total System Performance Assessment Viability Assessment is based on laboratory experiments using water with a more aggressive chemical composition than is realistic for Yucca Mountain. This approach results in radionuclides dissolving completely and quickly from the spent fuel (~ 400 years) and being released at a relatively high rate from the repository. The conservatism resulting from this approach leads to reliance on the waste package to limit any releases and has the potential to cause problems if unexpected phenomena occur or if changes in regulations require or permit changes in the amount or rate of radionuclides transported. A more definitive approach based on more complete knowledge could help reduce the conservatism.

Corrosion

Corrosion of the drip shield, the waste package, the spent fuel cladding, and drift structural materials such as the invert are key issues in evaluating radionuclide releases and determining chemical species produced. The fundamental mechanisms of specific corrosion processes that could adversely affect the lifetime of barrier materials must be better understood. The goal would be to enhance confidence (i.e., reduce uncertainty) in the required, very long-term extrapolations of corrosion data, which of necessity must be made from data taken over a much shorter time period.

Modeling

Elaborate and flexible models have been developed by DOE and NRC to analyze repository performance. These models are capable of handling a very wide range of chemical and other processes because of their modular nature and their built-in capability to incorporate new processes, new physical characteristics, and new data. However, the lack of some key supporting chemical data and of a firm understanding of some key chemical mechanisms may lead to questions about the adequacy and validity of the chemistry model calculations. This lack of key information in some areas can lead to the use of overly conservative assumptions and overly conservative results.

"Expert elicitation" (Kotra, et al., 1996), a highly structured and well-documented process whereby expert judgments are obtained, has been used to obtain values for use in the models where solid data are lacking, uncertainties are large, technical judgments are required about the conservative nature of bounding assumptions, or more than one conceptual model is consistent with available data. It is recognized that this approach, although valuable when lack of time and experimental difficulties preclude obtaining the necessary data, should not be used when it is practical to obtain the necessary data experimentally (NUREG–1563).

The complexity and sophistication of the process level models require that they be abstracted into the TSPA model. Abstraction[2] of the models further complicates treatment and may inhibit full incorporation of important coupled effects. Consequently, there may be a loss of detail in the model calculations for the approximations of radionuclide releases from the repository, thus adding to the uncertainty in the resultant doses calculated for the critical group. Coupled processes are difficult to model and should receive careful attention in the abstraction process. The chemical processes in particular are difficult to model because of interactions of the many chemicals present and because of the influences of water composition, air, and temperature. Similar comments apply to both the NRC models and the DOE models.

Conclusions

Chemistry plays a critical role in the near-field performance of the repository. There is an important and unavoidable coupling of the effects of water, air, and temperature with the near-field chemistry of the repository; these effects are inextricably bound up with repository design. The present DOE repository design of the near field relies on corrosion resistance of the waste package and drip shield as major contributors to the multiple-barriers concept for in-depth defense. The case for meeting the multiple-barriers licensing requirement, which is a major element of the defense-in-depth licensing philosophy, may be improved by the deliberate introduction of certain chemical features in the repository near field. In particular, carefully chosen backfill materials and materials to fill the void space in the waste packages can chemically alter radionuclides such as ^{99}Tc and ^{237}Np such that they are transported much less rapidly out of the near field through sorption or precipitation reactions. In this way the contributions of ^{99}Tc and ^{237}Np to the radiation dose over the time of interest for licensing may be substantially reduced. Because of the potential significant contributions of deliberately added chemical features to the repository, NRC should anticipate their inclusion and be prepared to evaluate them.
Specific activities that staff may undertake or should continue include:

- Staff should conduct scoping calculations of the importance of backfill to modify the chemical environment and to act as an attenuating agent for released radionuclides. If it is determined from theses calculations that the use of backfill or the effects of corrosion products can have an important effect on performance then more realistic analysis would be required by the applicant.

- Staff's analysis of the potential importance of secondary uranium phases show significant reductions in calculated dose: therefore continued work on the role of secondary phases is essential to be prepared for possible inclusion of this phenomena in DOE's safety case. This work should not only include secondary uranium phases, but also other secondary phases (e.g., iron oxyhydroxide corrosion products if DOE decides to include this as an important element in its safety case) that are likely to form and may sequester or attenuate key radionuclides.

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[2] Abstraction is a process whereby computer models are simplified to make calculations less time consuming and less expensive, or in some instances even possible. Model detail is lost in the process of abstraction. This loss may be compensated for by doing "off-line" calculations that support and justify the simplifications.

- Staff needs to continue its work on understanding the rates and especially the mechanisms of corrosion of the drip shields, the waste packages, and the spent fuel cladding.

- Staff needs to continue and refine careful analysis of the role of coupled processes in repository performance as part of its development of review capabilities. Because of the complexity of the models and the abstraction of these models into the TSPA the key focus is to ensure that an important coupled process is not left out of the performance assessment.

Introduction

This paper presents and discusses selected aspects of the near-field chemistry in the proposed Yucca Mountain high-level waste (HLW) repository. Based on this discussion, key chemistry areas are identified, about which additional studies would help the Nuclear Regulatory Commission position itself better to evaluate the U.S. Department of Energy's (DOE's) license application (LA) and to respond to changes in repository design. Three important chemistry areas are focused on for possible additional study in the technical literature, in the laboratory, and in modeling. The three areas are: *chemical reactions in engineered barriers, solids formation and dissolution, and corrosion.* Water composition, air composition and circulation, and temperature are discussed in the context of their effects on chemistry and on coupled effects in the repository near field. The importance of process-level modeling of the three chemistry areas and the impact of the drift environment on modeling is discussed in terms of needs that should be addressed in the abstraction of process modeling into Total System Performance Assessment (TSPA) models.

This paper highlights and discusses what the author views as the most important chemistry study areas in the "near field"[3] that merit increased attention by NRC staff. A more thorough analysis of these areas could help NRC position itself better to evaluate DOE's TSPA for the site recommendation (SR) and LA for the repository. Such analyses would also help position NRC to accommodate future changes in DOE's design assumptions for the repository in a timely way. The discussion in this paper is restricted to chemistry in the near field because it is there that the greatest possibility exists for changes in materials and engineering features of the repository and, consequently, for reductions in uncertainty in repository performance. A number of excellent reviews of the literature on the near-field environment and the geochemistry of the proposed Yucca Mountain HLW repository are available (Angell, et al., 1996; Murphy and Pabalan, 1994).

Chemical reactions in engineered barriers can have a significant effect on the rate of release and transport of radionuclides in the near field. Three types of chemical reactions have the potential to be of importance, namely co-precipitation reactions that can incorporate important radionuclides into the crystal structure of other solid chemical phases, oxidation/reduction (redox) reactions that could lead to less mobile forms of key radionuclides such as technetium and neptunium, and radionuclide sorption-desorption reactions that could attenuate possible releases of key radionuclides. Taken together these processes could be used to reduce the uncertainties associated with the mobilization and transport of important radionuclides such as ^{99}Tc, ^{129}I, and ^{237}Np. For example, oxidative dissolution of spent fuel and the subsequent formation of secondary solid phases (e.g., higher valent uranium oxides) at the spent fuel-water interface could have a profound effect on the rate of release of radionuclides from the spent fuel by producing solids that can co-precipitate and/or adsorb radionuclides or by possibly limiting further oxidation of the fuel. This is an important area for further study (Pearcy, et.al., 1994).

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[3] In this report, definition of the "near field" is limited to the contents of the drift, including features such as drift supports and other construction materials, drift backfill, drip shields, waste packages and internals, cladding, spent fuel, and the water and air that enters the drift. Broader definitions of the near field often include the zone of rock affected by the thermal pulse from the waste.

The Role of Chemistry in Repository Licensing

Chemistry plays a critical role in both the near-field performance of the repository and the performance of the natural (geologic) barrier (Simmons, et al., 1995). As reported by the TSPA Peer Review Panel (1999), "the near-field geochemical environment is an important and complex part of the performance assessment," Both DOE and NRC recognize that, because of the complexity of this environment, performance assessments of the proposed repository at Yucca Mountain will always rely on many assumptions and abstractions about near-field chemistry in making the case for licensing the repository for high-level radioactive waste disposal. This includes assumed and/or calculated near-field chemistry parameters in computer modeling of processes that affect corrosion, radionuclide dissolution, and radionuclide transport. The NRC staff will need to be in a position to evaluate DOE's assumptions and abstractions when it considers the SR and LA.

The most important chemical processes in the near field are corrosion (of drip shields, waste packages and their supports, and spent fuel cladding), radionuclide dissolution (especially of irradiated uranium dioxide, which contains the transuranium elements and fission products of interest), secondary chemical phase formation, and radionuclide transport (especially of neptunium, technetium, and iodine) through backfill material that may be present. The NRC must understand these processes sufficiently to be able to evaluate the validity and relevance of chemical information and the analyses of the chemical processes DOE provides in support of its LA. DOE must make supportable arguments about the chemical processes to defend the reliance of its LA on the presence of multiple barriers in the repository, which is a major component of the defense-in depth philosophy required by NRC. It is also important for NRC to be able to analyze the implications (especially the degree of conservatism) of the reliance DOE has placed on "bounding cases"[4] in its analysis.

The continuing changes that DOE is making in repository design substantially complicate NRC's task of preparing to evaluate the repository LA. Some of these design changes could have a profound effect on the chemistry and performance of the repository, and these changes in the design may lead to considerable additional work on the part of NRC and the Center for Nuclear Waste Regulatory Analyses (CNWRA) in experimental programs and modeling studies. Furthermore, additional changes to the proposed design are anticipated prior to the LA. A key issue is how NRC should focus its resources to have sufficient depth and breadth to review the final LA with whatever new design features and enhancements that DOE may choose to include.

Computer modeling plays a central role in predicting repository performance. The success of the models (including process-level modeling and total systems performance modeling[5]) rests, in part, on a thorough

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[4] The term "bounding cases" is used to denote instances in which the largest or smallest value believed to be realizable is used to provide an upper or lower limit to a calculated value. This is a useful approach when a more realistic value cannot be calculated, but it can lead to excessive conservatism and tends to compromise the concept of defense in depth by obscuring the true contributions of individual barriers.

[5] Process-level models are chemical and physical models and associated codes of the basic phenomena and processes that are thought to be important to repository performance. Total systems models are abstracted from the process-level models to represent these key processes in a simplified manner so that the analyst can evaluate the overall functioning of the system and identify the aspects and features of the system that are most important to overall performance. Although in principle one could simply combine all the process models into a

understanding of the physical and chemical mechanisms and processes that control the degradation of engineered barriers (including spent fuel and other waste forms) and the mobilization and transport of radionuclides. Chemistry of the near field is a key factor in the models used to analyze repository performance. Several computer codes are used in the models[6] to calculate phase equilibria and the distribution of species. Current generation models are constructed in such a way that they are capable of handling greater sophistication and breadth in describing the chemical processes for Yucca Mountain than has been required of them to date. The models, however, do not include databases for some of the more complicated coupled chemical processes[7] involved in solids formation and in chemical reactions in the backfill. This absence is due, in part, to the lack of data required for analyses of these potentially important chemical reactions and their kinetics, and to the fact that backfill has not been considered seriously by DOE until recent design changes. It is also due to the difficulty of carrying out the modeling. A particularly important modeling area is model abstraction, wherein the simplifications inherent in the abstraction process may leave out or overly simplify important coupled chemical processes.

The Near-Field Environment and Coupled Processes

It is useful to consider the essential features of the major environmental contributors to the chemistry of the near field. In this way a context is provided to think about the chemical considerations related to chemical reactions in engineered barriers and solids formation and dissolution. The key components that control chemistry in the near field are water, materials that react with it, and gases. The temperature of the system plays a major role in determining the rates and extent of reactions that take place between the aqueous and gaseous phases in the near field and support materials, backfill, container materials, spent fuel cladding, and the waste forms. These reactions are highly coupled as thermal-hydrological chemical (THC) processes and cannot be properly treated as individual processes.

Water composition and reactions in the proposed repository drifts will have a major effect on the chemical processes involved in radionuclide dissolution and transport, engineered barriers, and corrosion. Yucca Mountain tuff contains about 10 percent water by volume and the current percolation of water (from surficial processes) into the repository is estimated to be from 1 to 10 millimeters per year. The water contains minor but important constituents such as dissolved oxygen; calcium, carbonate, bicarbonate, silicate and chloride ions; and others. Chloride ions could be very important in corrosion. Carbonate and silicate ions are likely components of interfacial solids that may form. Chemically reactive materials may also be added to the water through reactions of the water with backfill. The added reactive materials could affect either the formation of solids or the transport of radionuclides such as technetium and neptunium by changing their valence. They could also change the pH of the water. The pH and the oxidizing or reducing

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single total systems model, the resulting model and code would be too cumbersome to operate efficiently and the amount of time required to conduct calculations would be excessive. Many aspects of the process-level models are not important to repository performance (though they may play some role) and therefore the approach that both NRC and DOE have taken is to abstract and represent key processes within the total system model.

[6] Examples of such codes are EQ3/6, PHREEQE, SOLMINEQ.88, SOLVEQ, MINEQL, MINTEQ, and ECHEM.

[7] Coupled processes are those processes wherein two or more processes are interrelated and interact in a manner such that a change in one process causes changes in the others.

nature of the water and its solutes will greatly influence the chemistry of primary reactions such as corrosion and of secondary reactions such as formation of solid mineral phases and redox reactions.

Radiolysis of water will form hydroperoxide radicals, hydrogen peroxide, and other oxidizing (as well as reducing) species, all of which have the potential to change the valences of important radionuclides, e.g., Np and Tc, thereby radically changing their chemistries. Radiolysis is unlikely to have a major effect on overall repository performance at any significant distance from the waste form because gamma rays will be the main source of radiolysis, and the gamma dose will decrease rapidly (being due largely to ^{137}Cs decay) relative to the 10,000-year licensing period of the repository. The effect of radiolysis at the surface of the spent fuel or vitrified waste could, however, be of some importance because of the presence of alpha-induced chemical species. The formation of nitrate and nitrite ions by radiolysis of nitrogen in the presence of water could produce the respective acids (nitric and nitrous) which, if formed in sufficient amounts, could affect the rate of dissolution of the waste form.

Table 1 presents information on the J–13 well-water composition commonly used to represent the water that will enter the drifts. The appropriateness of this composition as representative of water in the near field will need to be confirmed.

Table 1. Approximate Composition of J -13 Well Water (DOE, 1998)
(pH = 6.8 -8.3)

Chemical Species	Molality x 10^4
SiO_2 (aq.)	95.00 - 11.40
Na^+	18.30 - 21.70
Ca^{2+}	2.90 - 3.70
HCO_3^-	1.93 - 2.34
Cl^-	1.78 - 2.37
K^+	1.00 - 1.40

Air entering and circulating in the repository will have a major influence on the humidity and overall oxidizing conditions of the near field. As long as air is circulating through the repository, and in the absence of a reducing backfill, it will be impractical to establish a near-field environment that is, on average, reducing. Oxygen in the air will have a great influence on the chemistry of the radionuclides of greatest interest (e.g., Np, Tc, I, and U), and will play a major role both in providing an oxidizing environment to corrode the drip shield, the waste canisters, and the spent fuel cladding and in oxidizing constituents of the spent fuel and vitrified waste. It could also react with chemically reactive backfill materials. The oxidizing effects will be due largely to oxygen in the air and dissolved in the water, but could also be due to radiolysis in the short term. Radiolysis of nitrogen from the air will produce oxides of nitrogen that may ultimately form nitric and nitrous acids by reaction with water. The higher valence states of uranium (+6 versus +4) are more stable and form complexes more soluble than the reduced form of uranium. The oxidizing environment also determines the valence states of radionuclides such as technetium and neptunium by oxidizing them to species (TcO_4^- and NpO_2^+) that form more soluble complexes with water and are more readily transported by water than the reduced species. However, even though the overall repository conditions are oxidizing while air is being circulated, there will be local regions that are chemically reducing in nature. This would be the case, for example, in the immediate vicinity of the inner stainless steel waste package container for spent fuel and the steel containers for the vitrified waste when they are corroding and producing ferrous ions, which are moderately strong reducing agents under some conditions.

Other constituents of air (e.g., carbon dioxide and water) will play a role in chemical reactions of the fuel material when fuel cladding fails, and could potentially have a role in bacterially produced phenomena. The carbon dioxide in the air is a potential reactant for the formation of soluble carbonate complexes of several actinide elements, notably uranium and neptunium. Uranyl ions, formed by oxidation of uranium dioxide to UO_3 followed by reaction with water, can react with carbonate ions to form the very stable and highly soluble uranyl tricarbonate complex ions, (Grenthe, et al., 1992) providing an excellent example of this type of complexation reaction. Neptunyl ions also form stable carbonate complex ions. Such ions would be soluble in and readily transported by water. However, maximum amounts of carbonate complexes are formed at pHs somewhat above the expected pH of the water in the repository.

Temperature has a dominant influence on the relative humidity and the presence or absence of liquid water in the repository. Temperature also greatly affects the mechanisms and kinetics of corrosion and dissolution, the transport of radionuclides, solid phase solubilities, and chemical equilibria throughout the repository. In general, both thermodynamic free energy (which quantifies chemical equilibria) and reaction rates are related in a positive and exponential way to temperature, which contributes to the great importance of temperature to chemistry.

DOE proposes in its EDA-II design concept to control the temperature of the repository in such a way that water is driven away from the drifts, condenses in the areas between the drifts, and percolates into rock layers below the waste. During the time of the thermal pulse, the temperature also has a major effect on the corrosion potential of the new waste package design.[8] If spent fuel is exposed to the circulating water and air under these conditions, chemical reaction rates and equilibria, solubilities of solid phases, and possibly sorption-desorption reactions in the backfill could be adversely affected. Inclusion of backfill in the drifts will also have an important effect on the temperature of the near field by insulating the waste packages. Placement of specific materials in the near field in conjunction with control of the thermal loading of the repository thus can have an important influence on the expected water composition and amount, and, consequently, the chemical reactions that take place.

Because of the strong influence of temperature on chemical (and other) aspects of the repository, it makes a major difference whether the repository is cooled or not. Even in a cooled repository, the temperatures inside the waste packages will be much higher than the temperatures in the walls of the drift. An unventilated repository will reach a temperature dictated by radioactive decay, repository design, waste package loading, areal loading of the repository, and time. If backfill is introduced, repository temperatures will rise because of the insulating effect of the backfill, as noted earlier. Data on the effects

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[8] In a limited temperature regime between about 85 and 100° C, the outer waste package layer (alloy-22) is more susceptible to localized (crevice) corrosion than at either higher or lower temperatures.

of temperature on solubilities and equilibria and on the kinetics of corrosion are needed for the model calculations of the chemistry of the near field. In some cases such data have been obtained and incorporated into the models, but in some potentially important cases, for example for silicates, such data are lacking.

The Effects of Repository Materials on Near-Field Chemistry

DOE has gone through a succession of major and minor repository design changes during and following the period of preparation of the TSPA–VA. These design changes have resulted in major changes in repository materials and consequently in the anticipated repository chemistry and its modeling. For example, DOE's most recent design calls for the emplacement of backfill around the waste canisters (Barrett, 1999). Although the primary motivating factors appear to be preventing rock-fall damage to the drip shields and physically diverting moisture away from the waste packages, backfill could be engineered to have chemical properties that can influence the near-field chemical environment in a positive way. From another perspective, engineering of the near-field environment should be done to avoid placing materials in the repository that could lead to enhanced uncertainties for radionuclide releases. For example, in the most recent DOE design, the use of concrete as a major support material in the drifts is largely eliminated. The goal is to reduce the uncertainties associated with high pH fluids that could result from water reacting with concrete, possibly leading to higher solubilities and lower sorption of some radionuclides in the near field. Changing placement of waste packages has a major effect on repository temperature profiles. Changing the proposed materials of construction of the drip shield and the location of materials of construction of the waste packages has had a major effect on the model predictions of rates and amounts of radionuclide releases for a 10,000-year period. So too has the change from an uncooled to a cooled (using forced circulation of air) repository for the 50 (or more) years prior to closing. Continuing changes in design make NRC's job of preparing to review the expected DOE repository LA more difficult. If the LA is to be addressed in a timely way, NRC either needs reliable advance information on the repository design so it can be prepared to make an independent assessment of the results of DOE's analysis of the Yucca Mountain repository and of the LA, or it needs to be prepared in a general way to accommodate to possible future changes in the repository design. The NRC staff has developed a flexible model and code (TPA) to deal with the changing DOE designs, but needs to anticipate increased DOE reliance on near-field chemistry in its long-term safety case.

The repository will contain both spent Light Water Reactor fuel and DOE high-level waste in the weight ratio of about nine to one. There will be relatively small amounts of miscellaneous DOE spent fuel derived from a variety of experimental studies over many years. Appendix A presents the proposed allocation of repository space to types of waste and the principal proposed Yucca Mountain repository design features at the time of writing this paper. Once the waste package and fuel cladding (or the canister holding the vitrified waste) have been breached, the spent fuel material or vitrified waste will be exposed to chemical attack by the water that reaches them. The composition of the water reaching the waste forms will not be the same as that of the water that entered the drift because of the chemical reactions the water will have undergone by corroding the waste packages and by reaction with other materials (e.g., backfill) it has contacted. The water will also have been altered in composition by evaporative concentration of dissolved constituents such as calcium, carbonate (and bicarbonate), silicate, and chloride ions. Other chemical species will be formed by radiolysis. These changes, depending on how extensive they are, have the potential of having a profound effect on the dissolution reactions of the water with the waste form and the vitrified defense high-level waste. Appendix B shows representative physical and chemical characteristics of Pressure Water Reactor (PWR) and Boiling Water Reactor (BWR) spent fuels and the effects of burnup on the composition of the fuel material. A listing of the radionuclides and curie content in typical PWR and BWR fuel assemblies is given in Appendix C. A more detailed discussion of the radionuclide attributes is presented in Appendix D.

The degraded repository materials through which radionuclide transport takes place can have a major influence on the mechanisms and rates of transport by retarding the movement of contaminants.[9] The most likely and most abundant corrosion products in the near field will be those from the iron in the repository. Because the five-centimeter-thick stainless steel inner shell of the waste package will be very close to the waste form and external to it, there is a very good possibility that any actinides and fission products leaving the spent fuel or vitrified waste will come into contact with ionic ferrous (Fe^{2+}) and ferric (Fe^{3+}) ions or with solid oxides or hydrated oxides of iron, which will tend to bind the actinides and fission products in their structures. However, because of the insolubility of iron hydroxides, the concentrations of ferrous and ferric ions will be low, and redox reactions will be limited. The NRC staff will need to determine the importance of this process if DOE decides to include retardation in corrosion products as an important element in its safety case.

The presently proposed Yucca Mountain repository design calls for the inclusion of backfill in the drifts, which could act as an engineered barrier. The term "backfill" is generally interpreted to mean any material other than waste packages and structural materials that is placed in the repository drifts such that it partially fills the drift. Currently DOE is primarily considering relatively unreactive materials such as sand or crushed tuff as backfill. The use of chemically active backfill materials could be a major simplifying factor in modeling, for example by effectively removing reactants such as carbonate and bicarbonate ions, by controlling pH, and by removing oxygen. A realistic analysis of the use and amount of backfill material required to remove the above reactants would aid in determining the practicality of the use of backfill in this way. Backfill material in the drifts offers the following important potential chemical advantages:

- The rate and extent of transport of radionuclides, especially of technetium, neptunium, and iodine, could be greatly reduced by introducing chemical backfill systems in which chemical sorption is strong, that is, in which the K_d[10] is large. In the cases of some radionuclides (e.g., technetium and neptunium) it might be necessary to change the chemical species, such as by changing the valence, to enable effective sorption. Both sorption and chemical change might be accomplished by reaction with the same backfill material, or a mixture of backfill materials might be used to obtain the desired chemical conditions.

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[9] The term "retarded transport" means that the rate and extent of radionuclide transport are diminished by some mechanism. Retarded transport may be facilitated by engineered barriers or possibly through the formation of corrosion products or secondary solid phases as the waste degrades.

[10] "K_d" is a term that gives a quantitative measure of the retention of a dissolved chemical species on a solid sorbing medium. It is usually defined as the ratio at equilibrium of the mass of the sorbed species per gram of a solid sorbing phase divided by the mass of dissolved species per milliliter of aqueous solution (e.g., milliliters/gram). K_d is thus a measure of the fraction of a chemical species sorbed onto a solid phase. Other measures of sorption have also been introduced that evaluate the fraction of a dissolved species sorbed per unit surface area of a sorbing medium. By introducing a measure of the rate of movement of water through the sorbing medium, the retarded rate of movement of the chemical species through the sorbing medium may be calculated.

- The dissolution rates of waste-related materials in the drifts (e.g., waste containers, spent fuel cladding, spent fuel material, and vitrified defense high-level waste and its containers) could be substantially slowed by saturating the dissolvent (the incident aqueous solutions) with the chemical species being dissolved, e.g., the uranium in the spent fuel.

- Solids formation could be enhanced by addition of chemical species that form interfacial solids to inhibit dissolution and release of radionuclides from the waste form. For example, calcium and silicate ions that could react with uranium to form analogs of natural insoluble minerals such as uranophane could be added.

A large number of potential backfill materials might be considered and studied for use in the repository drifts to control radionuclide transport. Some of these have already been considered by both DOE and NRC, as noted earlier. In the presence of oxidizable materials (i.e., reducing agents) in the backfill, water and air entering the drifts could be depleted in oxygen before they reach the waste package. If desired, carbon dioxide could also be removed by an appropriately chosen backfill material. The exiting solutions of radionuclides produced by water contacting the waste form could react with carefully chosen backfill material, or with material added to the invert structure beneath the waste packages, to slow the passage of the radionuclides out of the drift. Some possible backfill materials that have been proposed for waste repositories are listed in Appendix E. The list of backfill materials in Appendix E is meant to be suggestive of the large number of potential backfill materials that might be considered, some of which might have very beneficial effects on repository performance. Because pertechnetate, neptunyl, and iodine ions are among the most difficult in the repository to control chemically, and because they are also among the principal long-term contributors to health concerns at the Yucca Mountain repository boundary (currently 20 kilometers south of the repository itself), these elements are discussed below.

Formation and Dissolution of Secondary Solid Phases

The potential importance of the formation of solid phases at the interface between water and the spent fuel has been pointed out by Murphy and Codell (1998). Nonetheless, to date there has been limited effort spent by DOE or NRC either experimentally or in modeling studies to elucidate the chemical reactions leading to solids formation and their ability to limit the release of fission products and actinide elements from the waste form (i.e., spent fuel and vitrified DOE high-level waste) or to bind them in the solid phases (Buck et al., 1998; Burns, Ewing, and Miller, 1997; Wronkiewicz et al., 1992). The great complexity of the chemical system, the effects of aging on the physical nature of the solids formed (e.g., many precipitates become more crystalline and their crystals grow and even change composition with the passage of time), and the kinetics of the solids formation processes pose formidable challenges to the experimentalist. These may be some of the reasons that little has been done in this area compared with the time and effort spent in other areas of repository performance.

One approach taken to addressing interfacial solids formation is to assume that naturally occurring uranium minerals will form or can be used as surrogates for interfacial solids that are likely to form (Apps, et al., 1993). This assumption is based on the fact that geological sites similar to the Yucca Mountain site have natural uranium minerals (e.g., uranophane, soddyite, and schoepite) that exhibit long-term stability. One such site that has been closely studied is the Peña Blanca site in Chihuahua, Mexico (Pearcy, et al., 1994). This site is in an arid and oxidizing environment with siliceous tuffaceous rock, much like Yucca Mountain. The presence of uranophane at this site suggests that uranophane is a mineral likely to form at the water-spent fuel interface in the proposed Yucca Mountain repository. The presence of calcium and silicate ions in the water in Yucca Mountain support this suggestion.[11] Even if uranophane is not formed, it is suggested that uranophane is similar enough to other minerals that might form that uranophane is a reasonable surrogate for those other minerals. This assumption, while reasonable, should be confirmed by experimental studies under conditions that duplicate as nearly as possible those at Yucca Mountain.

As the radionuclides in the waste packages are released, they will not necessarily be made available for immediate transport out of the near field. It is likely that some constituents of the water, such as silicate and calcium ions or carbonate ions, will react to produce solid phases of which the radionuclides are an essential part or in which they are bound up by processes such as sorption or inclusion. These reactions, perhaps augmented by other species (e.g., K and Na), will likely lead to precipitation of solid phases in the immediate vicinity of the waste form. As noted earlier, the solid phases could substantially impede the release of waste materials such as Np, Tc, Pu, and, possibly, I if those elements are present as suitable chemical species. The vitrified defense HLW waste is likely to form insoluble silicates with the incorporated waste materials because of the high silicate ion content. Better and more extensive experimental data on uranium solids formation and on the solubilities of solid species formed in the repository environment, especially of silicates, are needed. The NRC staff's analysis of the potential importance of secondary uranium phases shows significant reductions in calculated dose; therefore, continued work in this area is essential to be prepared for possible inclusion of this phenomena in DOE's safety analysis.

Corrosion

An understanding of the rates, and especially of the mechanisms, of corrosion of the drip shields, the waste packages, the spent fuel cladding, and the materials of construction of the drifts themselves (e.g., the waste package supports and the inverts) is fundamental to predicting the chemical behavior of the repository in both the short and long terms, and thus to determining the suitability of the repository for disposing of spent LWR fuel and DOE defense HLW and reactor fuels. The composition of the water, especially of the pH, incident on the drip shields and waste packages; the presence or absence of continuously replenished air; and the temperature of the repository are major determinants of corrosion rates, as are the materials of construction of the drip shields, waste packages, spent fuel cladding, and canisters of vitrified waste. The minor constituents of the water—especially chloride ions, but also other ions of a corrosive nature—that contact the waste packages and their contents will be important, especially if they are concentrated by evaporation of the water.

The effects of radiolysis and of microbial action on corrosion may be important and should continue to be studied to the point where their relative importance may be assessed. Radiolysis is likely to be more important to corrosion that takes place very near the waste form where alpha-particle-induced reactions are most likely. Microbiological processes (Pope et al., 1988), if significant, will be most prominent on the outer areas of the waste packages, where nutritional sources are most likely to be available for the bacteria.

Confidence in extrapolations of corrosion data to the very long times of applicability of the LA depends on an understanding of the mechanisms of corrosion. A great deal is known about the phenomenological nature of corrosion of the various materials in the drifts, but much less is known about the basic

[11] Uranophane has the chemical composition $Ca(UO_2)_2(SiO_3)_2(OH)_2 \cdot 5\ H_2O$. All of the chemical species necessary to form uranophane are likely to be present at the water-spent fuel interface.

mechanisms of corrosion. It is likely, of course, that it will not be possible to fully elucidate the mechanisms of some of the most important kinds of corrosion (e.g., of crevice corrosion and stress corrosion cracking of alloy C–22) by the time of the LA. Some of these types of corrosion have been studied for many years by many investigators with no consensus on their mechanisms. Nonetheless it is essential to obtain as much basic understanding as possible. It is especially important to learn as much as is practicable about *critical corrosion temperature regimes* for the materials of construction, the temperature range where corrosion is likely to be most severe. Additional confidence in the corrosion data may be obtained by adherence to American Society for Testing and Materials standards.

Materials of construction

Table 2 presents the amounts of metals per waste package in the present repository design concept. The values are very approximate and are presented to give an idea of the amounts of metals available for chemical reactions. They do not include things such as inverts, whose design is still uncertain.

Table 2. Approximate Average Kilograms of Metal per Waste Package Containing Spent Nuclear Fuel, Including Drip Shield*

Stainless Steel	Titanium	C–22 Alloy	Zirconium
10,000 kg	3,390 kg	2,390 kg	3,570 kg

* The values in the table are approximate. There are several sizes of waste packages and several possible waste package designs depending on the type of waste contained.

Corrosion may be either "dry" or "wet," depending on whether or not water is present. Water may be present either as bulk water or as a film of water caused by condensation of humidity in the air.

Reaction of oxygenated water with iron to produce, first, ferrous ions, and then, after oxidation of the metallic iron is complete, ferric ions, will add these ions to the water while removing oxygen from the water and the air until the iron is completely oxidized. The presence of this reducing environment will be both local and time-dependent. As already noted, the length of time reducing conditions can exist will depend on the amount and location of reducing agent in the drift (in this case, iron) and on the rate at which oxygen enters the drift in air and/or water.

The iron in the waste packages will not be effective in changing the chemistry of corrosion of the drip shields, except perhaps at the point of contact of the drip shield with the support, because the dissolved iron will leave the repository without contacting the drip shields. Transport of the radionuclides could, however, be affected because presumably they will leave the repository by the same route as the iron-containing water, and will contact it. Thus the radionuclides could in some cases be reduced, or react in other ways with the iron to form chemical compounds or other species with it. In any case there will probably not be enough iron in the drift from materials of construction to maintain a reducing environment over the 10,000-year licensing period. If a reducing environment can be maintained, it must be done by adding reducing material to the drift, presumably in backfill.

Drip shields

The drip shield, which is currently planned to be made of grade 7 titanium 1.5 centimeters thick, is subject to corrosion, as are the waste package and the fuel cladding. The drip shield will be effective in diverting bulk (liquid) water away from the waste package only as long as it remains intact. As the drip shield corrodes, it will introduce titanium ions into the water incident on the waste packages. Tri-valent titanium ions are good reducing agents. The drip shield may rest on the iron waste package support and thus be subject to enhanced corrosion. An ionic path could be established between the drip shield and the iron in the waste package supports. It is also possible that the reverse will happen. That is, corrosion of iron in the drift could produce ferrous and/or ferric ions that could provide an ionic path from the waste package to the titanium drip shield. It is not known whether either process will be of importance to the chemistry of the radionuclides, but it may affect the corrosion of the drip shield. The times involved are so long that these processes (the mechanisms) and their effects may be important and should be considered.

Waste packages

The current waste package design is shown in cross section Figure 1. The outer container has a 2 centimeter-thick wall of alloy C–22, whose composition is given in table 3. The inner container has a 5 centimeter-thick wall of stainless steel. It is expected that the stainless steel will corrode much faster than the C–22 once water breaches the C–22, and that in times far shorter than the repository licensing period it will be completely oxidized provided that the C–22 is in fact breached. As noted above, this steel could produce locally reducing conditions in the waste package or drift, and if oxidized, will add to the ionic iron content of the water. (Iron corrosion products would occupy more volume than the stainless steel from which they come, leading to potential adverse mechanical and structural effects. However, these effects are not considered here.) Unlike the steel in the waste package support and the steel invert, the iron ions from the stainless steel in the waste package container will be available for chemical reactions with the waste form itself at the water-waste interface.

There are extensive sources of information and data on the corrosion of high-nickel alloys, some of which are similar to C–22 (Betteridge, 1984; Schweitzer, 1996; INCO Inc., 1969 and 1995), but none for anything approaching the lengths of time being considered for the lifetime of the waste packages. The corrosion data that do exist for C–22 are very encouraging. However, pit, crevice, and stress corrosion are still matters of concern, and a better understanding of the mechanisms involved in these corrosion processes is needed before credit can be taken for the very long-term protection that DOE may postulate in its LA.

Figure 1. Waste Package and Drift Design Cross Section

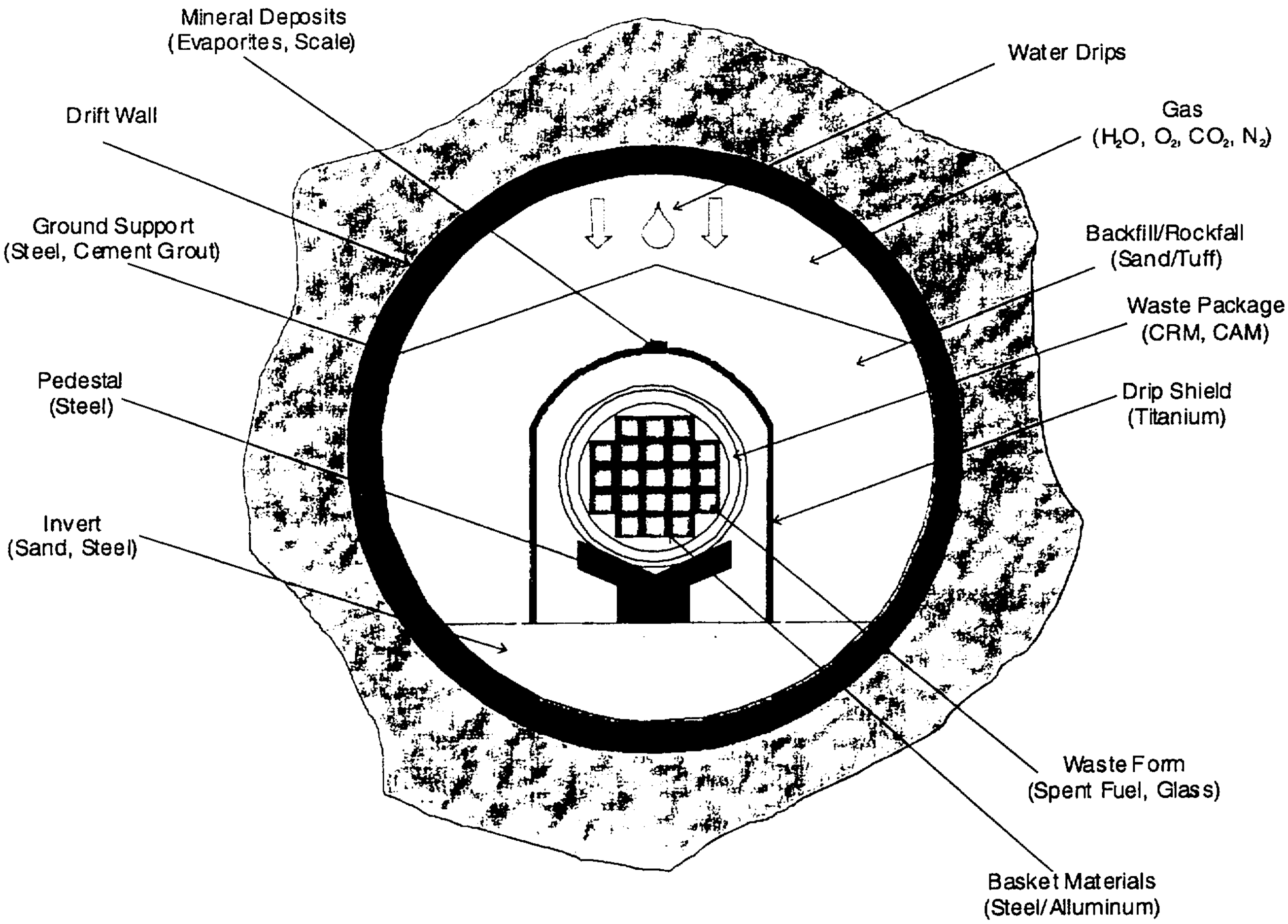

Table 3. Composition of Alloy C–22

Alloying Element	Weight Percent
Nickel	57.8
Chromium	21.4
Molybdenum	13.6
Iron	3.80
Tungsten	3.0
Vanadium	0.15
Manganese	0.12
Silicon	0.03
Carbon	0.004
Sulfur	0.002

The influence of temperature, and the possible existence of a temperature regime where corrosion is expected to be greatest, must be investigated. It is hypothesized, for example, that at temperatures above about 100° C, water will be driven away from the drift, thereby eliminating corrosion of C–22 for many hundreds or thousands of years. At temperatures below about 80° C, it is hypothesized that the rate of corrosion of C–22 will be so slow as to virtually eliminate corrosion. This hypothesis of a *critical temperature regime* illustrates the important role temperature plays in the chemistry of the repository. It also stresses the importance of repository design in determining temperatures, and thus chemistries, of important chemical processes such as corrosion.

Spent fuel cladding

Calculations based on corrosion data for the contribution of the Zircaloy cladding on commercial spent nuclear fuel to the prevention of the release of radionuclides from the fuel indicate a beneficial contribution by the cladding. (About 1.15 percent of the expected inventory of spent fuel in the repository is clad with stainless steel (DOE, 1998). All of this fuel is assumed in DOE's analysis to fail immediately.) A major question is how well the very good, but relatively short-term, corrosion data on Zircaloy can be extrapolated to the very long times (i.e., 10,000 years) required for the data to be valid for use in repository licensing. This is another example of the need for more understanding of the mechanisms of corrosion. The better the mechanisms are understood, the greater the confidence in both the data extrapolations and the multiple barrier arguments made in the LA. A potentially important factor that must be considered very carefully is the number of spent fuel elements that either already have failed before being put into the repository or are damaged during handling at the repository.

Vitrified defense high-level waste canisters

The canisters used for containing vitrified DOE defense HLW are made of steel. Once the waste package has failed, corrosion of the steel is expected to be relatively rapid as noted above for the stainless steel inner waste package container for the spent fuel. Like the iron from the stainless steel inner waste package container, the iron from the vitrified waste canister will be available for chemical reactions, primarily those involving chemical reduction(see Appendix F). In addition to the chemical reduction reactions caused by iron, iron oxides and hydrous iron oxides will sorb some chemical species. Such sorption may be an important factor in retarding radionuclide transport. The reactions of plutonium to form polymers are also addressed in Appendix F. These reactions, like the reduction of NpO_2^+ by iron, are also very dependent on pH.

Modeling

Quantitative evaluation of the ability of the Yucca Mountain repository to isolate radioactive wastes from the biosphere is a key component of the demonstration of the ability of the repository to meet the licensing requirements. This quantitative evaluation is achieved through the use of complex computer models with extensive databases.

Computer models exist that directly or indirectly minimize the free energy of a large number of interacting chemical systems. In this way the dominant chemical reactions in a suite of reactions are singled out. However, these models depend on databases that include all of the chemical important reactions, and are likely to be inadequate in some important aspects, for example, in solids forming reactions. The chemical processes that occur in the repository near field are extremely complex and are coupled, sometimes leading the modelers to take the conservative approach of assuming bounding cases.

Because of the complexity of the chemical nature of the repository and its contents, it is by no means certain that the bounding cases chosen are an adequate or reliable representation of the true situation. In fact, because of simplifying assumptions or inadequate or erroneous information, the bounding cases may not properly represent the facts, even in a conservative sense. Therefore, experimental verification of some of the results of computer analyses is necessary.

Model structure

The Yucca Mountain repository design is so complex that no one model is used to represent it. Rather, it is modeled through the use of multiple, interacting models. The Center for Nuclear Waste Regulatory Analysis has provided a literature review of coupled processes in the proposed repository (Manterfel, et al., 1993). DOE plans to synthesize technical and supporting information for its models through Process Model Reports (PMRs). At present the following nine PMRs are planned:

- Integrated site model.
- Unsaturated zone flow and transport model.
- Saturated zone flow and transport model.
- Near-field environment.
- Waste package degradation.
- Waste form degradation.
- Engineered Barrier System degradation and flow/transport model.

- Biosphere.
- Tectonics.

The models themselves are subdivided into components. Thus, the EBS PMR will be composed of:

- The physical and chemical environment model.
- The water distribution and removal model.
- The radionuclide transport model.
- The degradation mode analysis.

The physical and chemical environment model will have seven abstraction models. The chemical models are among the most complex and the most interactive (that is, coupled). It is these complexities that lead to the conservatism in the chemical aspects employed in the model analyses.

Model abstraction

In order to make practical calculations, it is necessary to "abstract" the complex models into a simplified form as noted above. This abstraction necessarily omits details of the repository near-field chemistry that have been judged by calculations, literature information, or technical judgment to be non-essential to obtaining valid results. In certain cases supplemental calculations are made to provide modeling information for the abstraction process. It is necessary to ensure that the various approaches to abstraction are indeed valid and that the omissions they entail are justifiable. This is especially true where technical judgment is the principal basis for the omissions or approximations.

Coupled effects

Water, air, and temperature govern chemistry in the near field. They can be controlled up to a point by the engineered barrier system; however, they are coupled and they cannot be controlled independently. Water in the drifts may be controlled, at least in part, by air flow and temperature. Control of water contacting the waste packages may also be achieved or attempted by diverting it from the waste package either by use of a drip shield or by backfill, for example by use of a "Richard"s Barrier (Conca, et al., 1998). Primary control of air volume may be obtained by having the repository open or closed. If it is open, air volume control may be either by natural or forced air circulation in the short term or by natural circulation in the long term. Temperature is controlled by repository design, that is, by arrangement and spacing of the drifts, waste package spacing in the drifts, design and content of waste packages, controlling circulation of air, and backfill. If backfill were not introduced for 300 years after closure of the repository (one scenario suggested by DOE) and the repository had been cooled until that time, then radioactive decay of the wastes would have reduced the production of decay heat to the point that the repository maximum temperature may not rise to levels that drive water from the drifts. Lowered maximum temperatures may permit consideration of a group of backfill materials (e.g., clays) that might otherwise be excluded because of alteration of their chemical and physical properties at higher temperatures.

Chemical effects are also coupled, both to each other and to water, air, and temperature. For example, solids formation at the waste form-water interface depends on presence of corrosion products, dissolution of the fuel form, constituents such as silicates dissolved in the water, the oxidation states of elements (as determined by oxygen in the air), and the temperature (which will determine the solubility). The interdependence and interactions of all these couplings lead to an extraordinarily complicated system that must be described, understood, and modeled.

Databases

Incompleteness of the thermodynamic and kinetic databases is a significant shortcoming of the chemistry models used to date for second-phase and mixed-phase formation, especially at the surface of the waste forms. The near-field chemistry, especially at the interface of water and fuel material or of water and vitrified waste, which is where second phases are most likely to form, is very complex. Many phases are possible in principle, and several are known to be likely under the Yucca Mountain repository conditions. Not only may their formation be slow, but the composition of the phases may change over time due to the well-known process of Ostwald ripening. Nonetheless, it is likely to be important to understand what phases do form, how soluble they are (especially as a function of temperature), and what their long-term behavior is. That is, how do they change with time? Do they become more refractory; do crystals grow larger; are additional elements incorporated into their structures over long time periods? Not all such data can be obtained before licensing the repository, but many can, and they are likely to be important in the licensing process.

Relatively little consideration is given in the databases to the possibility of valence changes or of new species formation brought about either by deliberate action or as a natural consequence of repository design. This is especially true in the cases of technetium, neptunium, and iodine. The paucity of data has been addressed in some instances by "expert elicitation." Expert elicitation is an accepted practice and must be resorted to when experimental data are not available and cannot be obtained in a practical sense (Kotra, et al., 1996). However, the divergence of opinions sometimes encountered among the experts of values to be assigned to chemical parameters attests to the uncertainties attending this approach. It is apparent that quality assurance of data in the usual sense cannot be obtained from expert elicitations.

Conclusion

Chemistry plays a critical role in the near-field performance of the repository. There is an important and unavoidable coupling of the effects of water, air, and temperature with the near-field chemistry of the repository; these effects are inextricably bound up with repository design. The present DOE repository design of the near field relies on corrosion resistance of the waste package and drip shield as major contributors to the multiple-barriers concept for defense in-depth. The case for meeting the multiple-barriers licensing requirement, which is a major element of the defense in-depth licensing philosophy, may be improved by the deliberate introduction of certain chemical features in the repository near field. In particular, carefully chosen backfill materials and materials to fill the void space in the waste packages can chemically alter radionuclides such as ^{99}Tc and ^{237}Np such that they are transported much less rapidly out of the near field through the action of sorption or precipitation reactions. In this way the contribution of ^{99}Tc and ^{237}Np to the radiation dose at the repository site boundary over the time of interest for licensing may be substantially reduced. Because of the potential significant contributions of deliberately added chemical features to the repository, the NRC should anticipate their inclusion and be prepared to evaluate them.

Specific activities that staff may undertake or should continue include:

- Staff should conduct scoping calculations of the importance of backfill to modify the chemical environment and to act as an attenuating agent for released radionuclides. If it is determined from these calculations that the use of backfill can have an important effect on performance, then more realistic analysis would be required by the applicant.

- Staff's analysis of the potential importance of secondary uranium phases shows significant reductions in calculated dose; therefore, continued work in the area of the role of secondary phases is essential to be prepared for possible inclusion of this phenomena in DOE's safety case. This work should not only include secondary uranium phases, but also other secondary phases (e.g., iron oxyhydroxide corrosion products if DOE decides to include this as an important element in its safety case) that are likely to form and may sequester or attenuate key radionuclides.

- Staff needs to continue their work on understanding the rates and, especially, the mechanisms of corrosion of the drip shields, the waste packages, and the spent fuel cladding.

- Staff needs to continue careful analysis of the role of coupled processes in repository performance as part of its development of review capabilities. Because of the complexity of the models and the abstraction of these models into the TSPA the key focus is to ensure that an important coupled process is not left out of the performance assessment.

Appendices

Appendix A. Repository Design Features

The repository is statutorily limited to 70,000 metric tonnes of uranium in nuclear fuel and other high-level waste expressed in terms of uranium and uranium equivalents[12] until a second repository is in operation. The proposed distribution of the uranium among the repository waste types is given in the following table .

Allocation of Repository Space to Types of Wastes

Waste type	Radioactivity, MCi	Metric tonnes U or U equivalent
Commercial spent nuclear fuel	19,000	63,000
Equivalent defense high-level waste	200	4,027
DOE-owned spent nuclear fuel	100	2,333
Equivalent commercial fuel	-	640

The present plan is that DOE-owned spent nuclear fuel will be placed in stainless steel canisters. The canisters will be placed in the center of waste packages surrounded by either 3 or 5 canisters of vitrified high-level waste. Some waste will be placed in small, high-integrity cans made of C–22 alloy prior to placement in the stainless steel canisters.

The details of the current repository design differ in significant ways from the design presented in the Viability Assessment (DOE, 1993. The following table gives a comparison of recent information on design features of the proposed repository with the VA reference design. This recent design is referred to as the Enhanced Design Alternative II (EDA–II).

[12] The amounts of wastes are expressed in terms of the amounts of uranium in the wastes. In the case of defense high-level waste, these amounts refer to the amounts of uranium that must have fissioned to produce the wastes, and are referred to as "equivalent" amounts of uranium.

Design Features of the EDA II Design

Criteria	EDA II	VA Reference
Repository area, acre	1060	740
Areal mass loading, MTU/acre	60	85
Drift spacing, meters	81	28
Drift diameter, meters	5.5	5.5
Emplacement drifts length, km	54	-
Access drift length, km	33.4	-
Ground support	Steel sets	Concrete lining
Invert	Steel with sand or gravel ballast	Concrete
Waste package materials	2 cm Alloy-22 over 5cm stainless steel	10 cm Carbon Steel under 2 cm alloy-22
Waste package capacity for PWR spent fuel assemblies	21	21
Waste package capacity for BWR spent fuel assemblies	44	44
Number of waste packages	10,039	-
Drip shield (placed at closure)	2 cm Ti-grade 7	None
Backfill (placed at closure)	Yes	No
Pre-closure period, years	50	50
Pre-closure ventilation rate, m^3/s	2-5	0.1
Temperature Cladding Drift wall Pillar center	 350° C 200° C 96 ° C	

Appendix B. Representative Characteristics of Spent LWR Fuels at Several Burnups

Attribute		PWR	BWR
Diameter/width	Fuel pellet Fuel rod Assembly	0.82 cm 0.95 cm 21.4 cm	1.06 cm 1.25 cm 13.9 cm
Fuel rods per assembly	Array Number	17X17 264	8X8 63
Height	Fuel stack Rod Assembly	3.66 m 3.85 m 4.06 m	3.76 m 4.06 m 4.47 m
Assembly weight		658 kg	320 kg
Fuel per assembly	Uranium metal Uranium dioxide	461 kg 523 kg	183 kg 208 kg
Metal hardware per assembly		135 kg	112 kg
Assembly volume		0.186 m^3	0.086 m^3
Avg. specific power, MW/Mg U		37.5	25.9
Burnup, Gwd/Mg U	Historical Future	33 60	27.5 46
Composition (historical burnup - future burnup) Initial ^{235}U enrichment, % Final Uranium, kg/Mg initial U Uranium enrichment, % ^{235}U Plutonium, kg/Mg initial U Other actinides, kg/Mg initial U Fission products, kg/Mg initial U		 3.30 - 4.73 955.4 - 922.2 0.84 - 0.54 9.47 - 14.38 0.71 - 1.8 34.4 - 61.6	 2.77 - 3.64 962.5 - 937.1 0.79- -0.57 8.26 - 12.3 0.59 - 1.50 28.6 - 49.1
Inventory (annual additions - cumulative), Mg initial U 1994 2000 2010 2020		 1207 - 19,024 1300 - 27,400 1400 - 39,000 700 - 50,200	 675 - 10.788 600 - 14,900 700 - 21,400 400 - 26,900

Appendix C. Typical Radionuclide Activities per Assembly for PWRs and BRWs.

(Taken from Draft Environmental Impact Statement for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye Country, Nevada, Volume II, Appendix A, page A-17.)

Especially important radionuclides from the point of view of repository safety are printed in bold face.

Radionuclide	PWR, curies/assembly*	BWR, curies/assembly**
Tritium	9.80e+01	3.4e+01
Carbon-14	6.4e-01	3.0e-01
Chlorine-36	5.4e-03	2.2e-03
Cobalt-60	1.5e+02	3.7e+01
Nickel-59	1.3e+00	3.5e-01
Nickel-63	1.8e+02	4.6e+01
Selenium-79	2.3e-01	7.9e-02
Krypton-85	9.3e+02	2.9e+02
Strontium-90	**2.1e+04**	**7.1e+03**
Zirconium-93	1.2e+00	4.8e-01
Niobium-93m	8.2e-01	3.5e-01
Niobium-94	5.8e-01	1.9e-02
Technetium-99	**7.1e+00**	**2.5e+00**
Rhodium-102	1.2e-03	2.8e-04
Ruthenium-106	4.8e-03	6.7e-04
Palladium-107	6.3e-02	2.4e-02
Tin-126	4.4e-01	1.5e-01
Iodine-129	**1.8e-02**	**6.3e-03**
Cesium-134	1.6e+01	3.4e+00
Cesium-135	2.5e-01	1.0e-01
Cesium-137	**3.1e+04**	**1.1e+04**
Samarium-151	1.9e+02	6.6e+01
Thorium-230	1.5e-04	5.8e-05

Uranium-232	1.9e-02	5.5e-03
Uranium-234	6.6e-01	2.4e-01
Uranium-235	**8.4e-03**	**3.0e-03**
Uranium-236	1.4e-01	4.8e-02
Uranium-238	**1.5e-01**	**6.2e-02**
Neptunium-237	**2.3e-01**	**7.3e-02**
Plutonium-238	1.7e+03	5.5e+02
Plutonium-239	**1.8e+02**	**6.3e+01**
Plutonium-240	2.7e+02	9.5e+01
Plutonium-241	2.0e+04	7.5e+03
Plutonium-242	9.9e-01	4.0e-01
Americium-241	1.7e+03	6.8e+02
Americium-242/242m	1.1e+01	4.6e+00
Americium-243	1.3e+01	4.9e+00
Curium-242	8.7e+00	3.8e+00
Curium-243	8.3e+00	3.1e+00
Curium-244	7.0e+02	2.5e+02
Curium-245	1.8e-01	6.3e-02
Curium-246	3.8e-02	1.3e-02

* Burnup: 39,560 MWd/MTHM; enrichment: 3.69 %; decay time: 25.9 yrs.
**Burnup: 32,240 MWd/MTHM; enrichment: 3.00 %; decay time: 27.2 yrs.

Appendix D - Radionuclide Attributes

Table 1. Radionuclides Important in the Repository

Radionuclide	Half life, yrs.	Type of radiation	Important oxidation states	Important chemical species
^{99}Tc	2.13E05	β	+7	TcO_4^- Tc_2O_7 Tc_2S_7
			+4	TcO_2 $TcO_2 \cdot H_2O$ TcS_2
			0	Tc°
^{129}I	1.57E07	β, e-, x-rays	+5	IO_3^-
			0	I_2
			-1	I^-
$^{238,235}U$	4.468E09 (^{238}U)		+6	UO_2^{2+} UO_3 $UO_2(OH)_2$ $UO_2(CO_3)_3^{4+}$
			5 1/3	U_3O_8
			+5	UO_2^+
			+4	UO_2
^{237}Np	2.14E06	α	+6	NpO_2^{2+}
			+5	NpO_2^+ NpO_2OH $NpO_2CO_3^-$
			+4	NpO_2 Np^{4+}
$^{239,240,241}Pu$	2.14E06 (^{239}Pu)	α	+6	PuO_2^{2+}
			+5	PuO_2^+
			+4	Pu^{4+} Pu colloid (Pu polymer)
			+3	Pu^{3+}

Spent fuel

The radionuclides in the spent fuel waste packages usually considered most important, either from the point of view of radiologic hazards or of chemical effects, are listed in Table 1 along with some of their radioactive properties, important oxidation states, and important chemical species. The principal constituent of spent LWR fuel is UO_2. It will have undergone physical and chemical changes due to fissioning. When LWR fuels are taken to the higher burnups (greater than 40,000 megawatt-days per ton) now being attained in some light water reactors these changes will become more extensive. (Manaktala, 1993) has provided a thorough discussion of the characteristics of spent nuclear fuel as the characteristics relate to source terms for radionuclides.

The early LWR spent fuel sent to storage will be the older fuel which has not experienced the high burnups expected for modern fuels. The Zircaloy-clad fuel rods are held together with Zircaloy tie rods attached to end pieces typically made of stainless steel. (Zircaloy is an alloy made of zirconium and small amounts of other metals). PWR fuel is expected to be the predominant fuel type in the repository. The weight ratio of PWR to BWR fuel assemblies in the repository is expected to be about 13 to 7. Each LWR spent fuel waste package will contain either 21 PWR or 44 BWR spent fuel assemblies. BWR fuel assemblies are Zircaloy-clad fuel rods enclosed in a sheet of Zircaloy channels along the length of the assembly. Therefore the ratio of the weight of metal hardware per assembly to the weight of fuel material per assembly is much higher for BWR fuel. The rods in both types of fuel are usually held in a square array by a Zircaloy grid spacer.

Vitrified defense high-level waste form

The principal constituent of the vitrified high-level defense waste is borosilicate glass which is used as an inert matrix to contain the defense high-level wastes. It typically holds about 25% by weight of the oxides of the waste. It is noteworthy that uranium and plutonium were removed to a large extent from this waste during the reprocessing operations, although neptunium was not. However, the very low burnup of the fuel in the reactor assures that the concentrations of actinide elements in the waste will be relatively low. The release of radionuclides from the glass is governed by the reactions of the glass itself. The temperature and composition of the water contacting the glass exert the major influence on the reactions that take place. The vitrified waste is a potential source of silicate ions in the repository. However, the already high SiO_2 content of the incident water should reduce the extent of attack on the waste glass. It is expected that a siliceous gel layer will form on the vitrified waste that becomes exposed to water, and that this layer will inhibit further attack of the glass and the concomitant release of radionuclides. However, it is not known how long this layer will last.

Transuranium elements

When a radionuclide has one or more of the following properties it becomes one of the important radionuclides in the repository: 1) high toxicity, 2) long half life, 3) facile transport through the environment, and 4) a high fission or neutron-capture yield. Because of the exceptional toxicity of the alpha-emitting actinides if they are taken internally, especially by breathing but also by ingestion, several of them, e.g., ^{237} Np and ^{239}Pu, are of primary importance from the point of view of health hazards.

Neptunium (^{237}Np) combines all four of the above properties and as a consequence is of importance in determining the licensability of the repository. The relative stability of the neptunyl ion (NpO_2^+) (except in redox reactions) is what permits its relatively rapid transport through media that retard many other ions. In the presence of oxygen or air, and in the absence of other reactants, neptunium is oxidized from lower valence states to the stable pentavalent state. Similarly, neptunium is oxidized to the pentavalent NpO_2^+ ion by nitric acid and nitrogen oxides (such as might be produced by radiolysis) when other redox agents are absent. Thus, it is apparent that the mobile NpO_2^+ ion will be the predominant neptunium ion in the repository in the absence of reducing agents.

Plutonium (^{239}Pu) also combines the above four properties that make it important in determining the licensability of the repository. Plutonium has the same valence states as neptunium and uranium, and has analogous oxygenated ions. However, the chemical stabilities of these states are quite different from those of neptunium and uranium. The redox potentials of plutonium are such that it is possible to have all four valence states existing simultaneously in aqueous solution. Although its pentavalent oxygenated ions are considerably more prone to disproportionation into hexavalent and tetravalent ions than the pentavalent state of neptunium, it is possible to have appreciable concentrations of PuO_2^+ in solution at low concentrations and low acidities, conditions that exist in the repository.

A Plutonium "polymer" forms readily when Pu(IV) hydrolysis products are heated and/or aged. Plutonium polymers are a colloidal form of plutonium that becomes less readily soluble in mineral acids upon aging and/or heating. Pu(IV) will be the predominant plutonium species in the repository in the absence of intentional changes to its valence. The importance of plutonium polymer derives from the fact that it does not behave as an ion, but instead behaves as a relatively large, electrically charged suspended particle whose behavior depends upon the medium it is suspended in (primarily on the ionic strength, but also on the presence of ions such as fluoride) and the solid surrounding that medium. Thus, its behavior is very difficult to predict (and to model) in the absence of quite detailed and complete information about its surroundings.

Uranium, though not as toxic as plutonium and not as much of a radiation hazard as neptunium, is included here because of the large amount of it in the spent fuel (as the reduced species UO_2). This makes it potentially important because of chemical reactions it may undergo. Important changes occur in UO_2 during fissioning. Restructuring of the UO_2 in irradiated fuel pellets is brought about by the fission reactions and the concomitant high temperature. Restructuring results in shattering the UO_2 and producing fractures that facilitate movement of volatile elements such as iodine, cesium and noble gases, as well as providing pathways for movement and agglomeration of the noble metals to produce metallic particulates. The fission product oxides occupy more volume than the uranium dioxide from which they are produced and this too leads to changes in the structure of the UO_2.

UO_2 in spent fuel will likely be attacked by water, as shown by the Nopal 1 deposit at Peña Blanca, Chihuahua, Mexico. If the uranium is oxidized to the hexavalent state, then carbon dioxide dissolved in the water will produce the highly soluble uranyl tricarbonate complex ion. As the uranyl tricarbonate complex ion is removed from the fuel material, the actinides and fission products will become accessible to chemical attack by the incident water and the dissolved constituents it contains, as well as by any radiolytically produced nitrogen-containing acids or other reactive radiolytic species. As already noted, uranium may react with ions such as silicate and calcium to form interfacial solids.

Fission products

Most of the fission products have half lives so short that they are of little consequence in the waste repository licensing analysis. In particular, ^{137}Cs and ^{90}Sr, which are the major contributors to radioactivity in the spent fuel and vitrified defense high-level waste during the first hundred years or so, will have radioactively decayed to negligible concentrations in much less than 10,000 years, although there will be some residual cesium present as ^{135}Cs. The only circumstance in which cesium and strontium might be important health hazards is their early release through breaching of waste containers by accidental or very unlikely natural causes, e.g., volcanism. There are, however, several fission product radionuclides that remain of concern in the very long term. Unfortunately they are isotopes of elements whose chemistries are extremely complex, notably technetium (Puigdomenech and Bruno, 1995) and iodine. These elements pose difficult problems in their chemical modeling and in control of their transport.

Technetium (^{99}Tc) is a fission product and a beta particle emitter. It combines the same four properties as ^{237}Np and ^{239}Pu and is one of the principal radionuclides of concern in the repository. Because of technetium's tendency to be heptavalent in the presence of oxygen it can form the anionic pertechnetate ion TcO_4^-. This ion, like the closely similar perchlorate ion (ClO_4^-), reacts very little to form complex ions or precipitates, and is only slightly sorbed by normally sorptive media such as clays. Its extremely long half life makes it one of the few fission product radionuclides that might contribute to a potential health hazard more than 1000 years after repository closure. Technetium can have several valence states, as pointed out in Table 2. It is highly desirable to find a practical way to change the valence from +7 to a lower valence, probably +4, that is more tractable chemically.

It is known that in LWR fuel a significant fraction of the technetium as well as of other noble metal elements are present as finely divided metals or oxides that resist dissolution even under very vigorous chemical treatment (de Regge, et al., 1980). Table 2 shows typical data for the dissolution behavior of some noble metals in irradiated fast reactor fuels (in which their concentrations are typically higher than in spent LWR fuels) under treatment with concentrated nitric acid. Although the temperatures and fuel burnups in fast rectors are higher than in LWRs, and consequently the amounts and extent of migration of the noble metals are greater, there will be insoluble metallic inclusions of them in LWR fuels. This is especially true in light of the progress toward higher burnups in LWR fuels (burnups up to 60,000 Mwd/tonne are anticipated), and the fact that only about one-third of the spent fuel expected to be disposed of in the repository is in the present, lower burnup, inventory. It is to be expected that the much milder conditions in the repository will produce virtually no reaction with the very refractory residues in LWR fuels. An implication of this fact is that the technetium radiation dose at the site boundary (20 kilometers south of the repository) may be as much as one-third less than that calculated in the current modeling studies, even if no chemical reactions are invoked to change the chemical species of the technetium. A similar situation exists for technetium in vitrified defense high-level waste (Sombret, 1999).

Table 2. Insoluble Metallic Inclusions in Fast Reactor Fuel Dissolution Residues*, wt. %
(Average of 10 experiments)

Tc	Ru	Mo	Rh	Pd
33.7	37.1	18.9	7.6	2.7

* Dissolver solutions: 10M HNO_3 followed by 10M HNO_3+0.1M HF

Iodine (^{129}I) also combines the four properties noted above, and is of potential concern because if ingested it concentrates in the human thyroid gland where it undergoes radioactive decay and is potentially harmful. The chemistry of iodine is very complex (Rudin and Garcia, 1992). It may exhibit all seven valence states, including the negatively charged iodide ion and elemental iodine. The fact that iodine is very volatile in its elemental state and that it forms few highly stable, refractory compounds makes its behavior particularly difficult to predict and to control and to model. In spent fuel it is usually found as the iodide ion, presumably in combination with cesium (with which it can combine because of the mobility of both iodine and cesium in the spent fuel). Because of the multiplicity of its valence states there are a number of possibilities for iodine to react with elements in the spent fuel as well as with materials within the repository and beyond it. Consequently it is not unreasonable to expect iodine to be retarded somewhat in its transport. However, it is known to move readily through soil. Experiments under actual repository conditions are necessary to determine how much retardation might be expected to occur. It is not out of the question to consider reacting and thus immobilizing iodine with a backfill material (such as a copper compound), but a careful study would have to be made of the reactions of iodine (and of copper) with other chemical species present to see if adverse chemical reactions might take place.

Appendix E. Partial List of Potential Backfill Materials

Tuff

The simplest backfill material to obtain and use is probably the material Yucca Mountain is made of, namely tuff. Tuff exists in several major types and may have zeolitic clays in it. Studies suggest that plutonium sorbs strongly on tuffs, but that neptunium and uranium do not. Technetium is also poorly sorbed. Not much is known about either the species likely to be present or the sorption behavior of iodine in tuff (although iodide ion is probably the species most likely to be present).

UO_2

It has been suggested (Forsberg, 1999and 1997), that the approximately 700,000 tonnes of depleted uranium stored as UF_6 at the uranium enrichment gaseous diffusion plants in the U.S. be converted to UO_2 and used as backfill in the drifts and/or waste packages as a way of disposing of the UF_6. Used in the drifts the UO_2 would saturate the water entering the drifts with uranium compounds, for example with uranyl tricarbonate after oxidation of the UO_2, thus potentially slowing dissolution of the spent fuel waste form, which is itself greater than 95 % UO_2. Adding uranium might also increase the rate and amount of secondary phase formation. It might also be advantageous to introduce the UO_2 around the spent fuel in the waste package. As mentioned earlier, in this way it could potentially react with the calcium and silicate ions in the water directly at the spent fuel-water interface to form a synthetic uranophane or similar natural uranium mineral analog and thus block the escape of radionuclides from the spent fuel. Although it may be premature for NRC to conduct an experimental program on UO_2 as a backfill in the absence of its study by DOE, it may be worthwhile to conduct a literature study of its potential use because of the advantages it appears to hold promise of offering.

MgO

Magnesia (MgO) has some potential advantages as a backfill material (Bynum, et.al., 1998). It could react with CO_2 entering the drift to form $MgCO_3$, thus removing or diminishing the possibility of carbonate and bicarbonate ion reactions with uranyl ions (and possibly with neptunyl ions) from the spent fuel to form highly soluble and mobile carbonate complexes. The rate and extent of dissolution of the spent fuel waste form through soluble carbonate complex ion formation would thus be reduced. The relatively small amount of CO_2 in the incoming air could be sorbed by MgO backfill.

Transition elements

There are several elements in the transition element series of the chemical periodic table that are potential drift backfill material or additions to backfill material. In general they are available as compounds in chemically reduced states that could react with oxygen entering the repository. In addition, they could react with the pertechnetate ions and neptunyl ions to produce lower-valent species that would be transported much less readily, either because of formation of precipitates or because of sorption-desorption reactions. Several such elements are discussed very briefly below.

Iron - The drifts already contain large amounts of iron in the form of materials of construction and in the waste packages. Therefore, iron is an obvious element to consider for inclusion in the backfill. Iron can form hematite which sorbs actinides strongly. In addition, iron hydroxides and oxyhydroxide (e.g., FeOOH), which would form under the repository conditions, are widely used as scavengers for a broad spectrum of ions in aqueous solutions.

However, the insolubility of Fe^{2+} hydroxide is so large that the reduced availability of ferrous ions and the problems of pH control are obstacles to its use for maintaining a reducing environment. Also, the importance of iron in bacterial processes introduces uncertainties into an already very complex chemical environment.

Manganese - Manganese is a potential reductant that might be added to the drift backfill. It has the advantages that the Mn^{2+} ion is a good reducing agent, and that Mn^{4+}, the likely product of the oxidation reaction, forms solid MnO_2 which, like ferric hydroxide, is a good scavenger for ions in solution.

Copper - Copper is another illustration of a possible reducing and scavenging agent. If cuprous sulfide were used in the backfill material then the possibility exists that both technetium and iodine might be sequestered, technetium as the highly insoluble sulfide and iodine as insoluble copper iodide. Introduction of sulfur into the drifts is problematical because of the possibility of bacterial action on the sulfur. However, copper is a bactericide so the extent of the problem would depend on the specific bacteria present.

For a reducing material to be useful in slowing the movement of radionuclides by changing their valences, the ratio of oxygen entering the drift to reducing agent present would need to be small enough that the radionuclides would be chemically reduced over a long enough period to be effective.

Potential waste package filler materials

Rather than leaving to chance the formation of interfacial solids such as uranophane by reaction of aqueous solutions of calcium and silicate ions in the incoming water with the UO_2 in the spent fuel, it is possible to place solids forming chemicals directly in the waste packages. This would increase the likelihood of solids formation, and would provide some control over the composition of the solids formed. It would also provide the possibility of adding chemical reactants such as reductants or precipitants for radionuclides such as technetium and neptunium. The reactants would be in the immediate vicinity of radionuclides as they left the spent fuel, and the likelihood of desired reactions taking place would be greatly increased. UO_2 is an example of a chemical that might be used to fill the void space around the spent fuel in the waste package. It could serve to saturate the water with uranium compounds and thus serve as a reactant to form a precipitate with ions such as silicate. It could also diminish the rate of dissolution of the spent fuel (which is largely UO_2) because the water would already be saturated with uranium. Similarly, reducing agents could be added to the waste package to reduce technetium and neptunium, and perhaps put iodine in an appropriate valence state for its controlled behavior.

Appendix F. Illustrative Chemical Reactions

Iron

The presence of iron in the drifts, both as a material of construction and as a major constituent of the waste packages, makes a discussion of its reducing properties instructive. The reduction of neptunium is discussed below to illustrate some of the important aspects of the reactions of iron. Some of the redox and precipitation reactions of iron are given below to provide a basis for the discussion.

$Fe^{\circ} \rightleftharpoons Fe^{2+} + 2e^{-}$

$Fe^{2+} \rightleftharpoons Fe^{3+} + e^{-}$ $\qquad E^{\circ} = -0.771$ v

$Fe^{2+} + H_2O \rightleftharpoons Fe(OH)^{+} + H^{+}$

$Fe^{3+} + H_2O \rightleftharpoons Fe(OH)^{2+} + H^{+}$

$Fe(OH)^{2+} + H_2O \rightleftharpoons Fe(OH)_2^{+} H^{+}$

$[Fe(II)*(OH)^2]_{ppt}$ $\qquad K_{sp/Fe(II)} = 1.64x10^{-14}$

$[Fe(III)*(OH)^3]_{ppt}$ $\qquad K_{sp/Fe(III)} = 1.1x10^{-36}$

Neptunium Reduction

It is desirable to keep or to put neptunium in the tetravalent state in order to slow its transport to the Yucca Mountain site boundary. The following simplified analysis suggests that iron may not be an effective reducing agent for neptunium.

$NpO_2^{+} + 4H^{+} + e^{-} \rightleftharpoons Np^{4+} + 2H_2O$ $\qquad E^{\circ} = +0.750$ v

$NpO_2^{+} + e^{-} \rightleftharpoons Np^{4+}$

The reaction of iron with neptunium is given by the equilibrium equation

$$NpO_2^{+} + Fe^{2+} + 4H^{+} \rightleftharpoons Np^{4+} + Fe^{3+} + 2H_2O$$

Using the values of standard potentials and solubility product constants given above, along with the usual relationship

$$\ln K = nFE^{\circ}/RT$$

it is found that, very approximately, at 25 °C,

$$(Np^{4+})/(NpO_2^{+}) = 6.4x10^{7}(H^{+})^{3}$$

According to the above calculation until the pH reaches about 2 there will be essentially no reduction of neptunium. Admittedly this is a crude calculation, ignoring as it does possible competing reactions as well as equating chemical activities to concentrations and using a simplistic value for ferrous hydroxide solubility. Nonetheless, it does point out the strong influence that the acidity (pH) may have on the equilibrium and consequently on the fraction of the neptunium reduced.

Plutonium

Plutonium is likely to be of concern in the repository because of the considerable tendency of Pu^{4+} to form plutonium colloids (often referred to as plutonium polymer). It is generally accepted that Pu polymer can form under repository conditions. Plutonium will almost certainly be tetravalent, and will hydrolyze to form polymer. Pu polymer is formed by the elimination of water from the hydroxide of Pu^{4+} and the formation of oxygen bridges. The redox and polymer forming reactions are presented below in simplified form.

$$PuO_2^{2+} + 2e^- + 4H^+ \rightleftharpoons Pu^{4+} + 2H_2O$$
$$PuO_2^{2+} + e^- \rightleftharpoons PuO_2^+$$
$$PuO_2^+ + e^- \rightleftharpoons Pu^{4+}$$

$$Pu^{4+} + 3H_2O \rightleftharpoons Pu(OH)_3^+ + 3H^+$$
$$Pu(OH)_3^+ + H_2O \rightleftharpoons Pu(OH)_4 + H^+$$
$$nPu(OH)_4 \rightleftharpoons nPuO(OH)_2 + nH_2O \qquad \text{(Polymer)}$$

References

Angell, P., J.A. Apps, G.A. Cragnolino, S.M. Hsiung, P.C. Lichtner, W.M. Murphy, R.T. Pabalan, D.A. Pickett, N. Sridhar, S.A. Stothoff, and D.R. Turner, June 1996. "Evolution of the Near-Field Environment in the Proposed High-Level Repository at Yucca Mountain—A Review of Hypothesis." ed. W.M. Murphy, San Antonio, TX: Center for Nuclear Waste Regulatory Analysis.

Apps, J.A., Bradbury, J.W., Cady, R.E., Ewing, R.C., Gustafson, D.L., Hofmann, R.B., Hoxie, D.T., Kovach, L.A., McNeil, M.B., Murphy, W.M., Ott, W.R., Pearcy, E.C., Sagar, B., Shea, M.E., Sridhar, N., Wittmeyer, G.W., Wood, J.R., Young, S.R.. September 1993. "The Role of Natural Analogs in Geologic Disposal of High-Level Nuclear Waste." eds. W.M. Murphy and L.A. Kovach: Center for Nuclear Regulatory Analysis–CNWRA 93-020.

Barrett, L.H., September 1999. Letter to J.L. Cohon, Chairman, U.S. Nuclear Waste Technical Review Board.

Betteridge, W 1984. *Nickel and its Alloys*. Chichester, UK: Ellis Howard.

Buck, E.C., R.J. Finch, P.A. Finn, and J.K. Bates. 1998. "Retention of Neptunium in Uranyl Alteration Phases Formed During Spent Fuel Corrosion. Scientific Basis for Nuclear Waste Management XXI," *In Materials Research Society Symposium Proceedings*, Davos, Switzerland, September 28–October 3, 1997, Vol. 506, pp. 87–94. I.G. McKinley and C. McCombie, eds. Warrendale, Pennsylvania: Materials Research Society.

Burns, P.C., R.C. Ewing, and M.L. Miller. 1997. "Incorporation mechanisms of actinide elements into the structures of U^{+6} phases formed during the oxidation of spent nuclear fuel," *Journal of Nuclear Materials*, 245 (1), 1–9.

Bynum, R.V., et alia., 1998. *Identification and Evaluation of Appropriate Backfills for the Waste Isolation Pilot Plant (WIPP)*, in Bennett, et al. (eds.), International Workshop on the Uses of Backfill in Nuclear Waste Repositories, Carlsbad, New Mexico, R&D Technical Report P178.

Conca, James L., Michael J. Apted, Wei Zhou, Randolph C. Arthur, and John H. Kessler October 1998. "Flow Barrier System for Long-Term High-Level Isolation: Experimental Results," Nuclear Technology, pp88–99, Vol. 124.

Department of Energy (DOE), December 1998. Viability Assessment of a Repository at Yucca Mountain, Vol.3. Total System Performance Assessment, p 3-100. DOE/RW-0508/V3.

de Regge, P., D. Huys, J. Ketels, L. Vandevelde, and L. H. Baetsle. 1980. Dissolution of Mechanically Mixed UO_2 -PuO_2 and Insoluble Residue Characteristic, Fast reactor fuel reprocessing, Proceedings of a symposium sponsored by the Society of Chemical Industry and held at UKAEA Dounreay Nuclear Power Development Establishment, 15-18 May 1979, London Society of Chemical Industry.

Fairhurst, C. August 9, 1999,"Engineering Barriers at Yucca Mountain—Some Impressions and Suggestions," in letter from B. John Garrick, Chairman, Advisory Committee on Nuclear Waste, to Greta Joy Dicus, Chairman, Nuclear Regulatory Commission. "Comments on DOE's License Application Design Selection Process (LADS) and Recommended Repository Design."

Forsberg, C.W. 1997. Long-Term Criticality Control in Radioactive Waste Disposal Facilities Using Depleted Uranium. Proceedings of the Topical Meeting on Criticality Safety Challenges in the Next Decade, Chelan, Washington, September 7–11, 1997. LaGrange Park, IL. American Nuclear Society, Inc.

Forsberg, C.W. February 1999. "Depleted-Uranium-Dioxide Particulate Fill of Spent Nuclear Waste Packages." Submitted to Nuclear Technology, unpublished, Manuscript Number 9375.

Grenthe, I., J. Fuger, R.J.M. Konings, R.J. Lemire, A.B. Miller, C. Nguyen-Trung, and H. Wanner. 1992. *Chemical Thermodynamics of Uranium.* Amsterdam, Holland: North-Holland Publishing Company.

INCO Inc. 1969. Engineering Properties of Nickel-Chromium Alloy 610 and Related Casting Alloys Birmingham, UK: Nickel Development Institute.

INCO Inc. 1995. High-Temperature, High-Strength Nickel Base Alloys, Birmingham, UK: Nickel Development Institute.

Kotra, J.P., M.P. Lee, N.A. Eisenberg, and A.R. DeWispelare. November 1996. *Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.* NUREG–1563.

Manaktala, H.K. May 1993. Characteristics of Spent Nuclear Fuel and Cladding Relevant to High-Level Waste Source Term, CNWRA 93-006.

Manteufel, R.D., M.P. Ahola, D.R. Turner, and A.H. Chowdhury. July 1993. A Literature Review of Coupled Thermal-Hydrologic Mechanical Chemical Processes Pertient to the Proposed High-Level Nuclear Waste Repository at Yucca Mountain, NUREG/CR–6021, CNWRA92–011,

Murphy, W.M. and R.T. Pabalan. November 1994. *Geochemical Investigations, Related to the Yucca Mountain Environment and Potential Nuclear Waste Repository.* NUREG/CR–6288.

P.A. Schweitzer, ed 1996. Corrosion Engineering Handbook. Marcel Dekker, Inc., New York, NY.

Pearcy, E.C., J.D. Pirkyrl, W.M. Murphy, and B.W. Leslie. 1994. "Alternation of uraninite from the Nopal I deposit, Peña Blanca district, Chihuahua, Mexico compared to degradation of spent fuel in the proposed U.S. high-level nuclear waste repository at Yucca Mountain, Nevada." *Applied Geochemistry 9* (6), 713–732.

Pope, D.H., et al., 1998. Microbiologically Influenced Corrosion: A State of the Art Review, Second Edition, MTI Publication No. 13. *Materials Technology Institute of the Chemical Processes Industries, Inc.*

Puigdomenech, Ignasi, and Jordi Bruno. April 1995. A Thermodynamic data Base for Tc to Calculate equilibrium solubilities at temperature up to 300°C, SKB Technical Report 95–09.

Rudin, M.J., and R.S. Garcia. February 1992. *National Low-Level Waste Management Program Radionuclide Report Series*, Volume 4: Iodine–129. DOE/LLW–120.

Simmons, Ardyth M. February 1995. *The Critical Role of Geochemistry in the Program Approach, DOE White Paper on Geochemistry.*

Sombret, P. 1999. personal communication.

Whipple, C. G., R.J. Budnitz, R.C. Ewing, D.W. Moeller, J.H. Payer, and P.A. Witherspoon. February 11, 1999. "Final Report, Total System Performance Assessment Peer Review Panel," prepared for Civilian Radioactive Waste Management and Operating Contractor, Las Vegas, NV.

Wronkiewicz, D.J., J.K. Bates, T.J. Gerding, E. Veleckis, and B.S. Tami. 1992. "Uranium release and secondary phase formation during unsaturated testing of UO_2 at 90°C.," Journal of Nuclear Materials, 190, 107–127.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

January 20, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: COMMENTS ON THE DRAFT ENVIRONMENTAL IMPACT STATEMENT FOR YUCCA MOUNTAIN

The Advisory Committee on Nuclear Waste (ACNW) received a copy of the Draft Environmental Impact Statement (DEIS) for the proposed Yucca Mountain repository in August 1999. We heard briefings by the Nuclear Regulatory Commission (NRC) staff on their preliminary review of the DEIS at our 114th meeting in November 1999, and from the Department of Energy (DOE) at our 111th and 115th meetings in July and December 1999, respectively. The Committee also had the benefit of comments on the DEIS from stakeholders during the 113th meeting held in Las Vegas, Nevada, in October 1999.

The Committee provides the following comment and recommendations:

1. The Committee remains concerned about the general unresolved issue of how comparisons and trade-offs should be made among real exposures in the near term and calculated exposures in the long term.

2. The Committee recommends that no additional work should be done to support the no-action scenario.

3. The Committee recommends that the final EIS should provide more detail on impacts and mitigation of alternative transportation scenarios.

<u>DISCUSSION</u>:

Radiological Impacts

The material in the DEIS with which the ACNW is most familiar is that related to the calculated long-term radiological effects following closure of the repository. The basis for the consideration of postclosure impacts in the DEIS is essentially DOE's Viability Assessment (VA). As DOE notes in the DEIS, "this EIS describes and evaluates the current preliminary design concept for the repository." The ACNW previously commented on the VA, and the NRC staff has conveyed to DOE the views of the ACNW as well as its own on issues that are unresolved in the VA. Any significant changes that are made in the postclosure analysis resulting from design changes will ultimately have to be reflected in the final EIS in a form appropriate for the National Environmental Policy Act (NEPA). In accordance with the Council on Environmental Quality (CEQ) requirement, Section 1502.9 requires that if any significant changes are made in the analyses because of design changes or otherwise, the changes have to be reflected in supplements to either the draft or the EIS. These changes have to be reviewed by the NRC staff.

The DEIS describes radiological impacts of pre-closure activities, as well as impacts following closure. The calculated impacts are based on the VA design, including the same thermal loadings. After the DEIS was issued, DOE has recently moved to a lower temperature design for the repository. There is a possibility that pre-closure exposures to radioactivity could increase under the low-temperature design as a result of increased handling of fuel, for example, if blending of fuel of different ages were required to control the heat load to the repository. Although we recognize that calculations can be made to satisfy the formal requirement of the NEPA in this instance, the ACNW remains concerned about the general unresolved issue of how comparisons and trade-offs should be made among real exposures in the near term (e.g., to workers from increased handling of fuel) and calculated exposures in the long term (e.g., to a hypothetical critical group 10,000 years in the future from ingestion of contaminated ground water). The DEIS is not the vehicle for resolving the issue, but we believe that such trade-offs should be explicitly made[1].

The No-Action Alternative

In the case of the DEIS for Yucca Mountain, the Nuclear Waste Policy Act specifically exempts DOE from having to present alternatives to geological disposal and alternative sites for a repository. DOE chose to include in the DEIS a no-action alternative. The no-action alternative consists of two scenarios intended to provide a baseline for comparison for the proposed alternative, which is described as construction, operation, and closure of a repository at Yucca Mountain. The no-action scenarios are open to criticism because of their lack of realism. In our opinion, there is no realistic "no-action" alternative for the long term. The realistic alternative is likely to be deferral of a decision on a repository for, say, 100 years. No-action in the sense considered by DOE in the DEIS (i.e., leaving fuel in dry-cask storage at reactors) may be of interest for 100 years, but it is not credible for 10,000 years. We believe that DOE may already have spent more effort than is worthwhile in analyzing the no-action scenarios. Effort spent on exploring more fully the site-specific analyses for Yucca Mountain would be a better investment than additional efforts spent on providing more detail for a 10,000-year no-action alternative.

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[1]We recognize that the question of how to handle issues of intergenerational equity is vexed (e.g., see P.R. Portney and J.P. Weyant 1999, **Discounting and Intergenerational Equity**, Resources for the Future, Washington, D.C.). Nevertheless, we think that explicit reporting of relatively certain exposures to radiation in the near term separate from highly uncertain exposures calculated for the long term would provide the information in the clearest format in the EIS.

Transportation

One of the major concerns expressed by stakeholders is that the transportation analyses in the DEIS are deficient. A main point of these concerns is that DOE failed to choose a preferred route and a preferred mode of transportation. DOE defends its decision to not specify mode and route on the basis that it is premature to select a route, and that they want public input to play a significant role in making a final choice. DOE believes that "the EIS provides the information necessary to make decisions regarding basic approaches" and that "follow-on implementing decisions, such as selection of a specific rail alignment, would require additional field surveys, state and local government consultation, environmental and engineering analyses, and NEPA reviews." DOE considered different options but not in detail; therefore, meaningful comparisons among the impacts and mitigation strategies of different options cannot be made. Hence, the proposed alternative of the DEIS is incomplete.

The ACNW sees the lack of detailed analyses of impacts and mitigation strategies, especially those stemming from the incomplete specification of transportation routes and modes, as a deficiency of the DEIS. We anticipate that the risks from radiological exposure[2] will be very small for any route, but we can envision the possibility of considerable differences among alternate routes and modes in terms of traffic risks, land-use impacts, and other items. The NEPA process is designed to expose impacts of alternative actions for projects that fall under the purview of the act and to present mitigation strategies for the alternatives so that valid comparisons can be made[3]. Thus, we conclude that the final EIS should provide more detail on impacts and mitigation of a transportation scenario and alternates to it.

Sincerely,

B. John Garrick
Chairman

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[2]DOE reports impacts of radiological exposure in the DEIS as latent cancer fatalities. These are calculated using the linear, no-threshold hypothesis (LNTH) in association with very small dose rates collectively to some target population. As we noted in our letter of June 4, 1999, on the LNTH, we think that expressing potential effects of very low doses, especially collective doses, in terms of cancer fatalities is a poor choice from a scientific perspective.

[3]Sections 1502.14 of the CEQ and Sections 102(2)(c)(i),(ii),(iv), and (v) of NEPA require that comparisons be supported by analyses.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

January 24, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: RUBBLIZATION — A DECOMMISSIONING OPTION

During its 114th meeting on November 17-19, 1999, the Nuclear Regulatory Commission (NRC) staff briefed the Advisory Committee on Nuclear Waste (ACNW) on the rubblization dismantlement approach to meeting the license termination rule. The subject was discussed with the Commission on December 15, 1999, in a public meeting. Rubblization is the sequence of operations whereby the above-grade parts of concrete structures are emptied and the partially decontaminated structures are demolished and disposed of in the intact and partially decontaminated parts of the structures that are below grade. After rubblization, the sub-surface material would be covered with fill material. A desired goal is to produce a site with unrestricted-use license termination and has no requirement for ongoing monitoring of radioactivity in the soil. This goal is interpreted to mean that the 25-mrem/yr requirement and as low as reasonably achievable (ALARA) principle have been met.

Recommendation

It is recommended that methods be developed for verifying radiation doses of rubblized sites. The primary requirement is that radioactivity in the bulk material be measured with sufficient accuracy to ensure that if the site is breached and the rubblized material is exposed, no one will receive a radiation dose greater than 25 mrem/yr. It is noted that the NRC Office of Nuclear Regulatory Research has commissioned two studies on how to measure radioactivity in bulk material.

The ACNW believes that rubblization may be a viable option, but there are numerous technical and policy issues. For instance, the method of measuring and monitoring residual radioactivity should be consistent with that used for other decontamination and decommissioning (D&D) and waste disposal activities. We, furthermore, suggest that in light of the projected cost savings, industry should take the lead in developing a basis for this process. The quantification of release levels for the applicable standard should be resolved early on. It is recognized that interagency discussions may be necessary. The ACNW will continue to interact with the NRC staff on this most significant issue.

Discussion

The nuclear power industry is giving serious consideration to rubblization of reactor containment and associated buildings. Maine Yankee Atomic Power Company, supported by the Nuclear Energy Institute, has engaged the NRC staff in discussions on application of such a concept to the decommissioning of Maine Yankee. In anticipation of future requests from industry to use rubblization for license termination, NRC is studying rubblization. The current level of effort by the NRC staff is minimal, but reasonable, as it waits to see to what extent the industry follows through with an application to use this process in decommissioning nuclear facilities. It is the current view of the NRC staff that existing regulations are adequate to provide a basis for evaluating license termination requests employing rubblization.

Although rubblization is now being considered with respect to decommissioning reactor buildings, it should be borne in mind that the concept may be extended to the decommissioning of other facilities.

Concerns have been expressed by government agencies, and public and special interest groups that rubblization may violate the ALARA principle or would not lead to unrestricted-use sites and may, in fact, lead to a proliferation of what are in essence low-level waste (LLW) disposal sites. Concerns have also been expressed that rubblization is contrary to the philosophy NRC has adopted in the past for disposing of radioactive wastes. It is the view of the ACNW that as presently conceived, rubblization has little in common with LLW disposal sites.

It is the view of the ACNW that the basic issue with rubblization as a method of unrestricted license termination is whether the NRC can reach a finding of reasonable assurance that rubblized sites meet the license termination requirements and are safe. This view derives from concerns about the applicability and use of methods currently accepted for estimating the radiation doses at rubblized sites. The nature of the rubblized material is such that evaluating its radioactive material content and doses from it, both at the site and in ground water, may prove to be difficult and expensive.

Structural steel reinforcing rods, as well as chemical elements in the concrete biological shield, may have become radioactive from neutron activation. Therefore, the resultant radioactive contamination may not only be surficial but also may be within the body of the rubble. In other cases there may be penetration of radioactive contaminants into cracks in the concrete resulting in internal contamination. This internal contamination will not be removed by commonly employed surface decontamination procedures such as scabbling[1] and sand blasting. Furthermore, and more importantly, the amount and extent of internal contamination may not be easily measured by the usual radiation survey techniques.

There will be a tradeoff between the hazards and costs associated with rubblization and the hazards and costs associated with removing the contaminated structural material offsite to an LLW burial site. In some cases, a combination of rubblization and removal offsite may prove to

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[1] Scabbling is a decontamination process whereby a few millimeters, up to several centimeters, of the concrete surface are removed by mechanical means such as scraping or chipping.

be the best approach. ALARA considerations may play an important role in determining the best course of action.

Although the potential cost savings through rubblization could be considerable, they could be offset by requirements for extensive decontamination or for taking large numbers of samples and performing large numbers of analyses. Any application to support the use of the rubblization process should include a risk comparison for various options and a cost-benefit analysis.

The ACNW views the use of rubblization as a potentially attractive approach to license termination. However, it is essential that methods of measurement of radioactivity contained in the rubble be available to provide reasonable assurance that rubblized sites meet the license termination requirements and are safe. It is very important to study a rubblization test case to elucidate the problems and the potential approaches to their solution.

Sincerely,

B. John Garrick
Chairman

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

March 21, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REGULATORY APPROACHES FOR CONTROL OF SOLID MATERIALS (CLEARANCE RULE)

The Advisory Committee on Nuclear Waste (ACNW) received a briefing from the NRC staff on the status of Regulatory Approaches for Control of Solid Materials (Clearance Rule) on December 16, 1999. In addition, the Committee reviewed the NRC published Issues Paper [FRN (6/30/99)] on release for unrestricted use of solid materials with small amounts of radioactivity and has held public meetings on the subject. Additional meetings are planned.

Recommendations

On the basis of information received to date, the ACNW makes the following recommendations:

1. The ACNW recommends that any regulations about the control and release of radioactive contaminated materials be based on radiation dose rather than on precedent, such as might be derived from the exemption of coal ash from regulation.

2. The ACNW recommends that criteria be established that will provide a consistent and rational basis for regulating materials with similarly low levels of radioactivity. It is especially important that consistency be based on dose not radioactivity content of material because of self-shielding by the material.

Discussion

At the present time, there are neither NRC nor Environmental Protection Agency regulations for control of most slightly contaminated solids. NRC makes its decisions on a case-by-case basis as licensees seek to release solid materials. The lack of release criteria could lead to potentially inconsistent release levels, and consequently to nonuniform levels of protection.

Draft NUREG-1640, which is out for comment, provides guidance for clearance of materials and equipment but does not set criteria for a rule. It does, however, relate radioactivity on or in material or equipment to radiation dose. Steel, copper, aluminum, concrete, as well as equipment, were studied for potential reuse. An intrinsic problem in carrying out the measurements of radioactivity in these and other materials is the difficulty of measuring radioactivity at very low levels and, more specifically, measuring the levels within materials (because of self-shielding by the material) in addition to levels on their surfaces.

We believe that regulations should be based on dose. The fact that coal ash is exempted from regulation and that the dose from uranium and thorium and their daughters in the ash has been discussed as a possible precedent for setting a dose limit for other materials should not be used to provide guidance for regulation. The radiation exposure paths from ash are very different from the radiation exposure paths from many slightly contaminated materials, such as steel, aluminum, copper, and other metals.

International groups, such as the International Atomic Energy Agency and the Commission of European Communities, have suggested 1 mrem/year as an acceptably low dose limit. European countries may adopt this standard.

There is an obvious need for reliable and sufficiently accurate methods of measuring or calculating the radioactivity inside materials (i.e., volumetric versus areal). With materials such as metals, the radioactivity within the body of the material should be identified because there is the possibility that the material will be remelted and mixed with either fresh or recycled contaminated metals and thus bring internal radioactivity to the surface. During melting, slag may form that concentrates radioactivity, providing a potential mechanism for increased dose.

The Committee believes the outlined proposal by the NRC staff provides adequate protection of the public health and safety. Some representatives of the steel and scrap industries oppose unrestricted use of slightly contaminated materials on the grounds that consumers might not want such products. The concern might be justified on the basis of as low as reasonably achievable. The Committee does not believe that this is a safety issue. Considering the potentially large amounts of contaminated materials imported from countries that may currently allow recycle, it is difficult to see how use of contaminated steel, scrap, and other materials can be controlled in the long run.

Conclusion

The costs of segregating and disposing of slightly contaminated materials will still be large relative to the perceived health benefits obtained even if a suitably low permissible dose is chosen. Therefore, we believe that recycling and reuse of slightly contaminated material is a reasonable course of action, subject to adoption of a dose limit and rational and consistent criteria that address both national and international issues.

Sincerely,

B. John Garrick
Chairman

References:

1. U. S. Nuclear Regulatory Commission, NUREG-1640, Vols. 1 and 2, "Radiological Assessments for Clearance of Equipment and Materials From Nuclear Facilities," Draft Report for Comment, March 1999.
2. Nuclear Regulatory Commission, 10 CFR Part 20, Proposed Rules, "Release of Solid Materials at Licensed Facilities: Issues Paper, Scoping Process for Environmental Issues, and Notice of Public Meetings, *Federal Register*, Vol. 64, No. 125, June 30, 1999.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

March 31, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: COMMENTS AND RECOMMENDATIONS ON THE DRAFT FINAL RULE, 10 CFR PART 63, "DISPOSAL OF HIGH-LEVEL RADIOACTIVE WASTE IN A PROPOSED GEOLOGIC REPOSITORY AT YUCCA MOUNTAIN, NEVADA"

Dear Chairman Meserve:

In this letter we offer our comments on the draft final rule, 10 CFR Part 63, "Disposal of High-Level Radioactive Waste in a Proposed Geologic Repository at Yucca Mountain, Nevada," and the NRC staff draft final responses to public comments on several technical issues addressed in the draft rule (Reference 1). This letter responds to the staff requirements memorandum dated February 1, 2000, requesting the views of the Advisory Committee on Nuclear Waste (ACNW) on the draft final rule by March 31, 2000.

During the ACNW's 114th meeting on November 14–16, 1999, the NRC staff presented a summary of the public comments received on the proposed draft 10 CFR Part 63 and its interim proposed responses to the comments. In addition, during a joint Advisory Committee on Reactor Safeguards (ACRS) and ACNW meeting on January 13-14, 2000, on defense in depth, the NRC staff presented its proposed approach for clarifying the multiple-barrier requirement in the draft high-level radioactive waste (HLW) rule. Since that time, the staff has kept us informed of changes to the draft rule as it has evolved; however, the Committee has not reviewed the ultimate version of the draft final rule that will be forwarded to the Commission.

The following comments are submitted on the staff's proposed response to several specific technical issues addressed in the rule. These issues include the staff's proposed approach to clarify the multiple-barrier requirement and defense-in-depth concept, aspects of design basis events, waste retrievability, human intrusion, performance confirmation, and transportation of HLW.

RECOMMENDATIONS

1. Multiple Barriers — The Committee recommends that a quantitative dose limit not be set in the rule for hypothetical assessments for performance of multiple barriers. The Committee recommends an approach to quantify the contributions of barriers that compares estimated repository performance with and without the benefit of specific barriers. The Committee recommends that the detailed method of analyzing multiple

barriers be limited to the guidance documents, as opposed to being a basic part of the regulation.

2. Performance Confirmation — The Committee agrees with the staff on the need for a repository performance confirmation plan that provides insights on post-closure performance while not compromising design flexibility. The Committee recommends that in its review of the Department of Energy's (DOE's) performance confirmation plan, the NRC staff encourages DOE to design its monitoring program in parallel with the repository design. Optimal placement of monitoring devices should not be precluded.

3. Design Basis Events —The ACNW supports the staff's proposed clarifications, including elimination of the term "design basis event" in the proposed rule to avoid confusion and miscommunication. The Committee recommends that the importance of event sequences in terms of their impact on overall repository performance, that is, on the radiation dose to the critical group, be the principal basis for allocating analysis and investigation resources to ensure the safety of the public and protection of the environment.

4. Human Intrusion — The Committee recommends that the staff avoid the use of surrogate risk values for human intrusion in the regulation. We recommend that the staff compare the results of the hypothetical intruder analyses to the results of the performance assessment analyses. If the staff decides that a license application could be evaluated more easily with a comparison of the results of the hypothetical calculation with a higher dose limit, for example, 1000 mrem (10 mSv) per year, we recommend that this approach be incorporated into the guidance rather than the rule itself.

5. Waste Retrievability — The Committee supports the staff's proposed approach to require DOE to plan for but not to demonstrate that the waste package is retrievable before issuing a license to construct the repository. The Committee believes that waste retrieval does not present an insurmountable technological challenge.

6. Transportation — The ACNW supports the staff's decision not to address transportation in 10 CFR Part 63. The Committee continues to emphasize the need for clarification and improved management of the overall transportation issue as no single agency appears to have the authority to take a total systems approach to address this public policy issue.

General Comments

1. The staff has done an outstanding job of summarizing and responding to the vast number and wide range of public comments received on the proposed draft rule. The ACNW commends the staff for this significant and noteworthy effort.

2. The staff has made considerable progress in its goal of improving public involvement during the past year through its interactions with the public on draft 10 CFR Part 63. We commend the staff for holding multiple workshops in the Yucca Mountain area to solicit input from stakeholders on the proposed rule. We also encourage the staff in its plans to hold follow-on workshops with stakeholders to convey the final resolution and

response to the public comments and its plans to post comment resolution on the Internet.

Specific Comments

1. Multiple Barriers

We understand that the staff's approach in the proposed regulation for demonstrating multiple barriers is to require that DOE demonstrate reliance on both natural and engineered barriers and that the repository system not depend unduly on any single barrier. We understand that the staff plans to require use of hypothetical calculations wherein barriers are assumed to perform to a lesser degree than anticipated, as a way of gaining insights into the contributions of barriers to overall repository performance. In addition, the staff may require in the rule that the results of the barrier underperformance analyses be compared to a numerical dose failure criterion. The staff also plans to provide more detailed guidance on acceptable methods to demonstrate compliance of multiple barriers in the Yucca Mountain Review Plan (YMRP).

The ACNW has closely followed the development of draft 10 CFR Part 63. In past advice, the Committee has endorsed the staff's general approach to address multiple barriers in the draft rule and has commended the staff for developing a regulation that captures the intent of risk-informed, performance-based (RIPB) regulation. We also advised the Commission that the performance of individual barriers should be quantified, and we recommended that the staff use a post-processor approach to decomposing overall repository performance assessments to quantitatively expose the contribution of individual barriers (References 2-6).

The Committee believes that the staff's proposal to calculate barrier underperformance is an acceptable approach for quantifying the contribution of individual barriers. However, we recommend that the staff not set a quantitative dose limit in the rule for comparison with the hypothetical assessments for performance of multiple barriers. In the spirit of a performance-based philosophy of regulation, the Committee would prefer that the measure of barrier performance always be in terms of its effect on overall repository performance. The ACNW recommends an approach (see enclosure) that involves comparison of risk curves showing calculated system performance with and without a specific functional barrier. Such an approach avoids comparison of the hypothetical results to a surrogate risk value or a subsystem requirement and, in our view, is more consistent with the staff's original performance-based strategy for draft 10 CFR Part 63 in SECY-97-300 (Reference 7).

We appreciate the competing demands placed on the staff to both specify a clear, numerical limit for evaluating compliance while at the same time develop a truly RIPB regulation that is less prescriptive. If the staff elects to use a surrogate risk value, as we understand is being proposed, we recommend incorporating the quantitative dose limits for the hypothetical calculations in the YMRP rather than in the rule itself.

2. Performance Confirmation

We understand that the NRC staff agrees with the public comments that some sections of the rule were too prescriptive and has modified the rule to allow DOE greater flexibility to develop a focused and effective performance confirmation plan. The Committee supports the staff's proposed approach to performance confirmation. We recommend that in its review of the DOE's performance confirmation plan, the staff encourage that DOE's monitoring scheme be designed in parallel with the repository design. Optimal placement of monitoring devices should not be precluded.

3. Design Basis Events

The staff is considering a number of clarifications in the proposed final rule, including eliminating the term "design basis event" and replacing it with the term "event sequence," to clarify that the probability of a design basis event is based on the entire event sequence.

The ACNW supports the staff's proposed clarifications, including elimination of the term "design basis event" in the proposed rule to avoid confusion and miscommunication. The Committee considers that the traditional concept of design basis is contrary to or at odds with an RIPB approach. The concept traditionally has been used to prescribe design requirements that are not necessarily linked to the performance measure of risk.

We recommend that the importance of event sequences in terms of their impact on the radiation dose to the critical group be the principal basis for allocating analysis and investigation resources to ensure the safety of the public and protection of the environment.

4. Human Intrusion

We understand that the staff is proposing to revise the consequence limit for evaluating human intrusion to an annual dose limit of 1000 mrem (10 mSv). This approach is consistent with the approach used in other NRC regulations for beyond-design-basis conditions. Other aspects of the hypothetical intruder analyses remain unchanged, that is, a single borehole is drilled at 100 years, a single canister is breached, and release of radionuclides to the groundwater pathway is evaluated. The staff believes that its proposed approach provides insights into the repository's resilience to human intrusion, yet avoids the undue conservatism that would result by comparing the results of the hypothetical intruder analyses to the overall performance objective of 25 mrem (0.25 mSv) per year.

The Committee supports the Academies' recommendation (Reference 8) pertaining to human intrusion to analyze different human intrusion scenarios for purposes of testing the robustness of the repository, not for calculating its probability of occurrence. We believe that the best approach to the human intrusion issue is to test the "hardness" of the repository and avoid debating arbitrary frequencies (for example, a 100-year drilling scenario frequency) for an event over which there is very little control.

The Committee recommends that the staff avoid the use of surrogate risk values, such as 1000 mrem (10 mSv) per year, in the regulation. We recommend that the staff compare the results of the hypothetical intruder analyses to the results of the performance assessment analyses. If the staff decides that a license application could be evaluated more easily by a comparison of the results of the hypothetical calculation with a higher dose limit, we recommend that this approach be incorporated into the guidance rather than the rule itself.

5. Waste Retrievability

The staff notes in its response to public comment that NRC will conduct an extensive and careful review of DOE's retrieval plans as part of any construction authorization review. However, DOE will not need to build full-scale prototypes at the time of construction authorization but will have to demonstrate technical feasibility of its retrieval plans using sophisticated computer simulations before receiving a license to receive and emplace waste. NRC notes that DOE needs to design and build the repository in such a way that the retrieval option is not rendered impractical or impossible. The staff proposes no changes to this section of the rule.

The Committee considers that waste retrieval does not present an insurmountable technological challenge. The Committee supports the staff's proposed approach to require DOE to plan for but not to demonstrate that the waste package is retrievable before issuing a license to construct the repository.

6. Transportation

In its response to public comments, the staff makes clear that transportation of HLW is not addressed in 10 CFR Part 63 because NRC and the Department of Transportation (DOT) have existing regulations that address transportation of HLW to a repository. The staff also offers some clarification of NRC's and DOT's role and governing regulations for the transportation of HLW in general and specifically to the proposed repository.

The ACNW has previously recommended that DOE be required to perform a comprehensive assessment of transportation risk to be evaluated by the NRC as part of the overall licensing decision regarding Yucca Mountain (Reference 9). A large experience base of the radiological risks associated with transportation already exists. The ACNW supports the staff's decision not to address transportation in 10 CFR Part 63. We continue to emphasize the need for clarification and improved management of the overall transportation issue because no single agency has the authority to take a total systems approach to address this public policy issue.

Sincerely,

B. John Garrick
Chairman

Enclosure:
B. John Garrick, Draft Technical Note, "On the Quantification of Defense in Depth," January 13, 2000.

References:
1. Memorandum (undated) from Donald A. Cool, Office of Nuclear Material Safety and Safeguards, to addressees, requesting review and concurrence on a Final Rulemaking

Establishing 10 CFR Part 63 - Disposal of High-Level Radioactive Waste in a Proposed Geological Repository at Yucca Mountain, Nevada (Predecisional).

2. ACNW letter dated October 31, 1997, from B. John Garrick, Chairman, ACNW, to Shirley Ann Jackson, Chairman, NRC, Subject: Recommendations Regarding the Implementation of the Defense-In-Depth Concept in the Revised 10 CFR Part 60.
3. ACNW letter dated October 31, 1997, from B. John Garrick, Chairman, ACNW, to Shirley Ann Jackson, Chairman, NRC, Subject: Application of Probabilistic Risk Assessment Methods to Performance Assessment in the NRC High-Level Waste Program.
4. ACNW letter dated March 6, 1998, from B. John Garrick, Chairman, ACNW, to Shirley Ann Jackson, Chairman, NRC, Subject: ACNW's Support for the NRC Staff's Approach to Assessing the Performance of Multiple Barriers.
5. ACNW letter dated July 29, 1998, from B. John Garrick, Chairman, ACNW, to Shirley Ann Jackson, Chairman, NRC, Subject: Comments on NRC's Total System Sensitivity Studies for the Proposed High-Level Radioactive Waste Repository at Yucca Mountain, Nevada.
6. ACNW letter dated September 3, 1998, from B. John Garrick, Chairman, ACNW, to Shirley Ann Jackson, Chairman, NRC, Subject: Advisory Committee on Nuclear Waste Comments on NRC's Draft 10 CFR Part 63 and Revision 0 of the Total System Performance Assessment Issue Resolution Status Report.
7. SECY-97-300, dated December 24, 1997, Subject: "Proposed Strategy for Development of Regulations Governing Disposal of High-level Radioactive Wastes in a Proposed Repository at Yucca Mountain, Nevada"
8. National Research Council, "Technical Bases for Yucca Mountain Standards," 1995.
9. ACNW letter dated January 20, 2000, from B. John Garrick, ACNW, Chairman, to Richard A. Meserve, Chairman, NRC, Subject: Comments on the Draft Environmental Impact Statement for Yucca Mountain.

Draft Technical Note

ON THE QUANTIFICATION OF DEFENSE IN DEPTH

B. John Garrick
January 13, 2000

PURPOSE

To propose a conceptual framework for quantifying the "defense-in-depth" aspects of the various levels of protection, provided in nuclear plants and nuclear waste repositories, against the release of radiation to the public and the environment.

GENERAL FEATURES OF THE APPROACH

The question is how can we best use probabilistic risk (performance) assessment (PRA and PPA) results to quantify and make visible the performance of the various "defense-in-depth" systems designed to provide multiple "levels of protection" against the release of radiation. Part of the answer lies in the way that the results are presented.

The key to the proposed approach, therefore, is a presentation format that clearly displays 1) the role that the individual safety systems play in providing protection against the release of radiation to the environment and 2) the effect of the individual systems acting in concert. This format allows for important risk and performance comparisons to be made at both the functional and system levels of a nuclear plant or a nuclear repository. It helps us make the important judgments of whether we are getting our money's worth from these multiple levels of defense, and whether we need more or less.

The approach utilizes the results of PRA and PPA. The scope of the PRAs and PPAs must include quantifications of information and modeling uncertainties, in the parameters used to measure risk or safety performance, and explicit identification of the supporting evidence on which these quantifications are based. The PRAs and PPAs must be structured in such a way as to reveal the process of assembling the results into the final measures of risk or performance, and to reveal the contributions, to these final measures, of the various levels of protection.

SPECIFIC FEATURES OF THE APPROACH

The answer to "how can we best use PRA and PPA results to quantify --- defense-in-depth ---" is believed effectively addressed using a two-dimensional structuring of risk and performance results. The structuring can be done in stages or phases in the spirit of a top-down approach. To illustrate the process at the functional level for reactors, consider Figure 1 with respect to the PRA of a boiling water reactor.

The rows of Figure 1 represent classes of initiating events at the functional level that can lead to core damage. In the first column (column 1) we plot probability curves showing our state of knowledge about the frequencies of the initiating events in the "probability of frequency" format. Columns 2—5 now represent the various safety functions that may respond to a particular class of initiating events. Column 6 contains the core damage frequencies for each class of initiating events. The sum of the Column 6 results represents the total core damage frequency, as illustrated in the last row.

The question is what entries should go in the boxes under the safety functions? The answer is to show the entries that best expose the defense-in-depth contributions of the safety functions. There are many possibilities. One possibility is to include three entries in each grid box, as shown in Figure 2.

As discussed further below, Entry 1 (Figure 2a) could be a probability curve indicating the unavailability frequency per demand of the safety function, given the particular class of initiating events. Entry 2 (Figure 2b) could be the core damage frequency, given the unavailability of the safety function, and Entry 3 (Figure 2c) could compare this result with the total core damage frequency of the last row. Doing this for each of the grid boxes would provide a clear perspective of the amount of protection provided by each of the functions. Different combinations of safety function availability and unavailability could be presented through the use of additional columns for making performance comparisons. Such analyses and comparisons provide a process for quantifying the role of various levels of protection, and hence, a quantification of contribution to defense-in-depth provided by different levels of protection.

TURNING UP THE MICROSCOPE

Now, the functional level shown in Figure 1 is too high a level to reveal performance characteristics of specific systems and barriers. To do that we need to turn up the microscope. Consider the grid box formed by the intersection of "Loss of Coolant" and "Inventory Control" of Figure 1. Suppose we detail that grid box into Figure 3.

Figure 3 divides the "Loss of Coolant" class of initiating events into six initiating event categories. It divides the "Inventory Control Systems" into eight more clearly defined protection systems. This level of detail is usually sufficient to provide quantitative engineering information on the levels of protection against exposing the public and the environment to radiation. The entries in the grid boxes can be the same as Figure 1 or modified as appropriate. In particular, Figure 2a indicates the unavailability of the safety system on demand, given the applicable initiating event. It reveals the reliability of the system under the conditions that the system is called on to operate and is the input used in the calculation of the core damage frequency for each specific category of initiating events. Figure 2b is the core damage frequency as a result of a particular category of initiating events, given the unavailability of the safety system (e.g., if that safety system were not present).

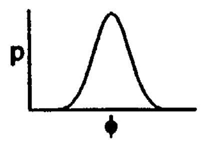
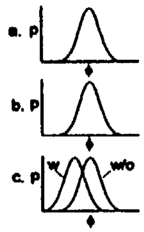
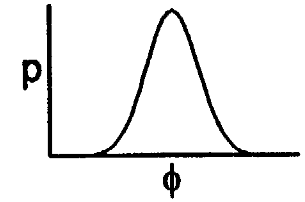
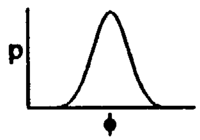
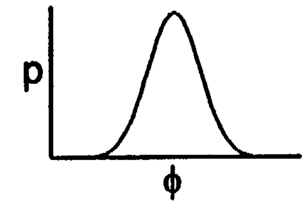
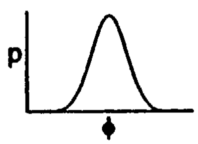
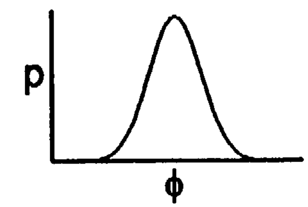
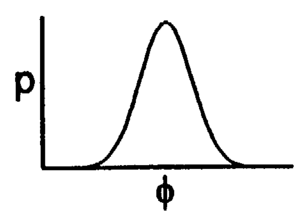
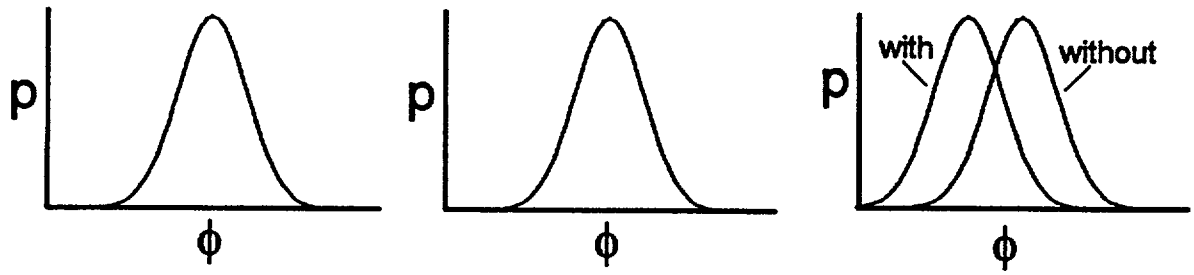
(1) Classes of Initiating Events	Safety Functions (2) Reactivity Control	 (3) Inventory Control	 (4) Heat Removal	 (5) Radionuclide Content	(6) Core Damage Frequency
Loss of Coolant 		Etc.			
Transients 					
External Events 					
Total Core Damage Frequency					=Σ(CDFs of Col. 6)

FIGURE 1. BWR SAFETY FUNCTIONS



System Unavailability Frequency per Demand
(a)

Core Damage Frequency (System Unavailable)
(b)

Total Core Damage Frequency (With and Without Safety System)
(c)

FIGURE 2. SYSTEM PERFORMANCE INDICATORS

Figure 2c is a key result in the quantification of the defense-in-depth of safety system protection. It is the total core damage frequency with and without the specific safety system being analyzed. It is important to note that Figure 2c is a different CDF than the one on which Figure 2b is based. The Figure 2b CDFs are those of Column 6. The Figure 2c CDF is the probabilistic sum of the Column 6 CDFs.

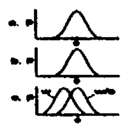
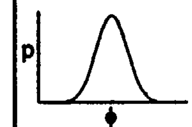
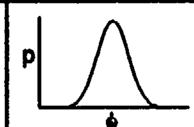
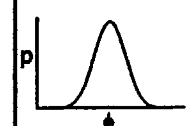
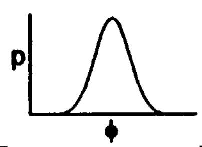
(Loss of Coolant Accident (LOCA) Indicators)	Safety Systems							Reactor Coolant System	Core Damage Frequencies (CDFs)
	Vessel-Level Makeup								
	Feedwater and Condensate	High-Pressure Core Spray	Reactor Core Isolation Cooling	Automatic Depressur-ization	Residual Heat Removal	Low-Pressure Core Spray	Fire Water		
Excessive LOCA		Etc.							
Large LOCA									
Small LOCA									
Breaks Outside Containment									
Interfacing System LOCA									
Other LOCAs									
CDF Due to LOCA Initiating Events				 					=Σ(CDFs of IE Categories)

FIGURE 3. BWR SAFETY SYSTEMS

APPLICATION TO NUCLEAR WASTE REPOSITORIES

Defense-in-depth of a nuclear waste repository takes the form of passive barriers whose performance must be analyzed over tens and hundreds of thousands of years. A two-dimensional display similar to the above can be constructed to exhibit the contributions of the levels of defense associated with a repository design. The functional barriers protecting the biosphere from radioactive contamination are, as shown in Figure 4, the spatial and flow control of water, the waste package containment, and the control of the mobilization and transport of radionuclides. The effectiveness of these barriers must be analyzed under a set of "geological scenarios" representing the possible climatological and geological events that might occur over tens and hundreds of thousands of years of the repository history. In Figure 4 these scenarios are represented in rows 2, 3, and 4. Row 1 represents the "base case" or "expected" scenario.

The point of Figure 4 is to display the contribution of the individual functional barriers to preventing the release of radioactivity to the biosphere. For this purpose we take, as the repository performance measure, the peak annual release to the biosphere, measured in curies.

In Figure 4, the rightmost column shows our state of knowledge about the peak annual release to the biosphere under the four geological scenarios. In the individual boxes of Figure 4 we display a pair of curves of the type shown in Figure 5. The curves show the contributions of the individual protective barriers by showing how the peak annual release would increase if that barrier were not present.

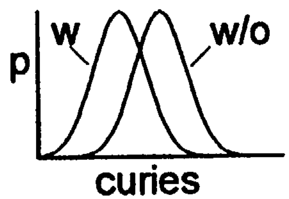
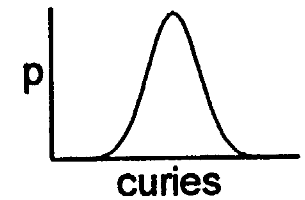
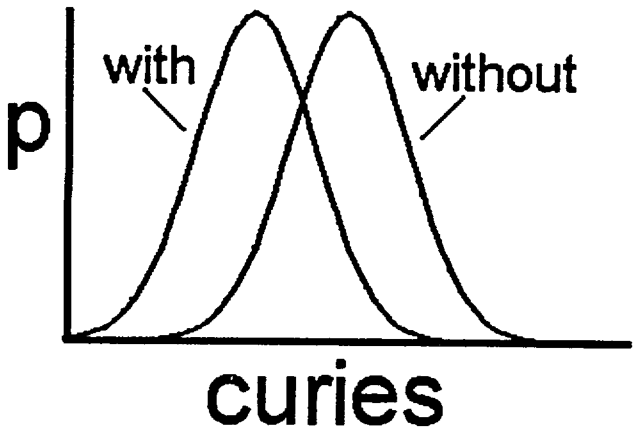
Initiating Conditions	Protective Barrier Functions			Peak Annual Release to the Biosphere (Curies)
	Water Flow and Spatial Control	Waste Package Containment	Radionuclide Mobility Control	
Current Climate		Etc.		
Geological Events				
Wet Climate				
Increased Geological Activity				

FIGURE 4. REPOSITORY PROTECTIVE BARRIER PERFORMANCE



Peak Annual Release
(With and Without Barrier)

FIGURE 5. PERFORMANCE COMPARISON

In Figure 6 we "turn up the microscope" on Figure 4 and recognize that the "barriers" shown in Figure 4 are actually composed of specific protective barriers. For example, the barrier "Water Flow and Spatial Control" of Figure 4 is now recognized as being composed of "Surface Runoff," which refers to a drainage system on the surface above the repository. Such a drainage system would divert the surface rainfall so as to prevent it from infiltrating into the ground above the repository. The column labeled "Water Diversion (Geotechnical)" refers to engineering the subsurface geology such as by the design of a Richards barrier. The column labeled "Water Diversion (Engineered Systems)" represents those engineered systems in the near field explicitly introduced to keep water from reaching the waste package. The rest of the columns are pretty much self-explanatory.

The individual boxes of Figure 6 show the impact of the protective barriers on repository performance by displaying what the peak annual release would be if that protective barrier were not present.

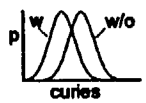
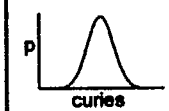
Initiating Conditions	Protective Barriers							Peak Annual Release to the Biosphere (Curies)
	Water Flow and Spatial Control Systems			Waste Package Containment		Radionuclide Mobility Control Systems		
	Surface Runoff	Water Diversion (Geological)	Water Diversion (Engineered Systems)	Corrosion Resistance	Fuel Cladding	Chemical Activities	Solubility, Retardation, Dilution	
Current Climate		Etc.						
Geological Events								
Wet Climate								
Increased Geological Events								

FIGURE 6. REPOSITORY PROTECTIVE BARRIER PERFORMANCE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

April 18, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

SUBJECT: ADVISORY COMMITTEE ON NUCLEAR WASTE 2000 ACTION PLAN AND PRIORITY ISSUES

Dear Chairman Meserve:

The Advisory Committee on Nuclear Waste (ACNW) has modified its 1999 action plan to update the priority issues it will consider in the year 2000 and beyond. We refer to the plan as an "action plan" rather than as a "strategic plan" to distinguish it from strategic plans required by the Government Performance and Results Act. A copy of the action plan is enclosed for your consideration.

The action plan is anchored to the NRC's Draft Strategic Plan for FY 2000–FY 2005 (NUREG-1614, Vol. 2) and supports NRC's mission, the principles of good regulation, and relevant strategies and performance goals identified by the agency. The plan is consistent with ACNW's revised charter and the ACNW's operating plan, which is being updated to reflect the priority issues identified herein.

One purpose of the ACNW action plan is to guide the Committee in carrying out its mission over the next year and beyond. The Committee identifies first-tier priority issues it will address this year and second-tier issues it will address if time and resources permit, unless directed otherwise by the Commission. In addition to the priority issues, the ACNW identified the process and report improvements that it will initiate this year to improve efficiency and effectiveness.

The Committee has identified five first-tier priority issues in this action plan, as follows.

1. **Site Suitability and License Application** reflect the increased activity associated with the proposed Yucca Mountain repository as the time for the site recommendation decision and the possible license application draws near.

2. **Risk-Informed, Performance-Based (RIPB) Regulatory Framework** acknowledges that the Committee remains committed and engaged in the agency's move toward an RIPB regulatory structure. Reviews of the agency's high-level waste regulation (10 CFR Part 63) and a fresh look at the defense-in-depth philosophy will fall under the RIPB issue.

3. **Decommissioning** will remain an area of continued focus. Topics that are included under this issue include various options for decommissioning nuclear power plants and materials sites, the Regulatory Approaches for Control of Solid Materials (Clearance Rule), and screening codes used to support decommissioning decisions.

4. The **Yucca Mountain Review Plan** will be closely monitored and reviewed as it is developed since it will contain the license application acceptance criteria.

5. **Transportation of radioactive waste** has become a first-tier priority issue in this year's action plan because of the public's interest and concern.

The Committee has also identified the following three second-tier priority issues in its action plan:

1. **Research** remains a second-tier priority issue, although the Committee will review waste-related research and report to the Commission on this topic.

2. **Low-Level Radioactive Waste (LLW) and Agreement States Program** remains a second-tier priority issue. The Committee continues to believe that the resolution of the LLW disposal problem (state compacts) is required to allow society to continue to benefit from the use of nuclear materials.

3. **Risk Harmonization** remains a second-tier priority issue. By risk harmonization, the Committee is suggesting that regulations within and across agencies reflect the same degree of protection for the relative risk posed by the hazard.

We would appreciate your comments or suggestions on the enclosed action plan.

Sincerely,

B. John Garrick
Chairman

Enclosure: As stated

THE ADVISORY COMMITTEE ON NUCLEAR WASTE
2000 ACTION PLAN AND PRIORITY ISSUES AND ACTIVITIES

This plan provides strategic direction to the Advisory Committee on Nuclear Waste (ACNW) in 2000 and beyond for focusing on issues most important to the NRC in carrying out its mission of protecting public health and safety, promoting the common defense and security, and protecting the environment. It also conveys ACNW's mission, vision, goals, and priority activities and indicates how these goals support the NRC's Strategic Plan.

SCOPE OF ACNW ACTIVITIES

The Committee reports to and advises the Commission on nuclear waste management. The bases of ACNW reviews include 10 CFR Parts 61, 63, 71, 72, and other applicable regulations and legislative mandates. The ACNW will undertake studies and activities related to the transportation, storage, and disposal of high-level and low-level radioactive waste (HLW and LLW, respectively), including the interim storage of spent nuclear fuel; materials safety; decommissioning; application of risk-informed, performance-based (RIPB) regulations; and evaluation of licensing documents, rules, regulatory guidance, and other issues as requested by the Commission. The Committee will interact with representatives of the public, NRC, the Advisory Committee on Reactor Safeguards (ACRS), other Federal agencies, State and local agencies, Indian Nations, and private, international, and other organizations as appropriate to fulfill its responsibilities.

OVERARCHING PHILOSOPHY

In conducting its self-assessments, the Committee realized that it has been the most effective in the areas in which it either initiated review of an issue or the Commission requested ACNW's advice on an issue. Examples include letters on the viability assessment, risk communication, and a white paper on repository design. The Committee has crafted its 2000 action plan to continue to place greater emphasis on self-initiated and Commission-requested topics.

The Committee will strive to take a top-down approach in its review of issues, focusing on the interconnection between issues and their crosscutting relationships, as opposed to reviewing issues in isolation.

The Committee also believes that it will best serve the Commission by taking an RIPB view in all of its activities. By this statement the Committee means that it will strive to ascertain the inherent risk associated with various issues, to encourage transparency in risk assessments, and to encourage consistency in the approach to risk assessments. The Committee will accomplish these goals by encouraging development of an overall flexible RIPB framework for materials and waste-related regulations. ACNW believes that an RIPB approach will remove or reduce rigid interpretation and prescriptive approaches in the application of regulations. The ACNW further believes that adoption of an RIPB framework could advance efforts toward risk harmonization and could alleviate conflicts associated with dual regulatory authority by providing a systematic and quantitative framework for assessing and comparing risk assessment approaches across and within agencies. An RIPB framework will allow for greater flexibility and transparency and will thus lead to greater confidence in regulatory decisions. In this way, the NRC can develop defensible regulations that have an obvious link to safety and can achieve a greater understanding of relative risks.

Finally, as part of its philosophy, the Committee aspires to factor in international experience whenever possible in examining issues. The ACNW also strives to consider creative ways to involve the public to a greater extent, including the decisionmaking process and ensuring that communication paths with the public remain open and effective.

ACNW MISSION

The ACNW's mission is to provide independent and timely technical advice on nuclear waste management issues to support the NRC in conducting an efficient and effective regulatory program that enables the Nation to use nuclear materials in a safe manner for civilian purposes.

ACNW VISION, DESIRED OUTCOMES, AND COMMITMENTS

In addition to a clear mission statement describing the ACNW's purpose, the Committee has identified a vision statement and desired outcomes to convey the Committee's direction, as well as commitments that guide the Committee toward these outcomes.

Vision

The ACNW strives to provide advice and to recommend solutions that are forward-looking, are based upon best available science and technology, can be implemented, and reflect the need to balance risk, benefit, and cost to society to enable the safe use of nuclear materials.

Desired Outcomes

The Committee aspires to achieve the following ultimate outcomes:

1. Advice is provided in adequate time to influence Commission decisions.

2. Advice is "forward-looking" in that it alerts the Commission to potential problems that may be averted by taking interim action, or forewarns the Commission of emerging issues that may require action at a later time.

3. Advice reflects state-of-the-art science and technology, yet is sufficiently practicable to allow for incorporation into NRC technical approaches, regulations, and guidance.

4. Advice for the intended audience is clear and concise.

5. Advice reflects an understanding of inherent risk and reflects consideration of the need to balance risk, cost, and benefit in all of NRC's decisions,

6. ACNW assists the Commission in making more transparent the regulatory decisionmaking process by operating in a spirit of openness and focusing on risk.

7. Advice identifies the interplay between HLW, LLW, and decontamination and decommissioning (D&D) programs wherever possible, as well as crosscutting relationships of issues to Environmental Protection Agency (EPA) and Department of Energy (DOE) programs.

8. ACNW is respected by the Commission, the NRC staff, EPA, DOE, and the public and is perceived as adding value.

9. ACNW is trusted by the public for providing frank, open advice and for offering a forum for public participation in the regulatory process.

10. ACNW assists in resolving conflicts between NRC and DOE, EPA, and other stakeholders by encouraging communication and providing a neutral forum for interaction.

Commitments

The Committee will carry out the following commitments in accomplishing its mission and in pursuing its desired outcomes:

1. Be responsive to the Commission's needs.

2. Challenge the status quo, as appropriate, thereby becoming an "engine for change."

3. Remain flexible, be responsive to change, and consider various options and contingencies.

4. Identify in advance those issues that could have an impact on NRC's ability to achieve its mission.

5. Focus on risk by asking "what is the risk, what are the contributors to risk, and what are the uncertainties?"

6. Be mindful of and begin to identify issues that cut across NRC waste and materials programs, as well as across EPA and DOE waste-related programs.

7. Foster an atmosphere of mutual problem-solving with the NRC staff.

8. Keep abreast of international trends and developments that could influence NRC policies or approaches, and factor international experience into Committee advice.

9. Consider the public as its ultimate stakeholder and seek improved approaches to obtain public involvement.

10. Maintain technical excellence and independence.

11. Abide by the Committee's action plan to ensure efficiency and effectiveness of Committee activities and products.

GOALS AND OBJECTIVES

The ACNW has developed general goals and objectives consistent with its mission and vision. The following five goals serve to provide strategic direction for the ACNW this year and support selected goals identified in NRC's Strategic Plan. For each goal, we identify objectives to help us better focus on our priority issues.

Goal 1: **Assist the NRC in positioning itself to respond to external change and uncertainty in the management of nuclear waste and materials. [This goal supports the NRC's Nuclear Waste Safety and Nuclear Materials Safety strategic arenas and NRC's strategic goal and primary Performance Goal to maintain safety, protection of the environment, and the common defense and security.]**

Objective 1: *Advise the Commission in a timely fashion on issues of a technical nature that may require changes in the regulations.*

Objective 2: *Inform the Commission about issues that could cause problems for the NRC or society if not given adequate attention, and recommend solutions.*

Goal 2: **Strive to ensure that NRC is employing the best science in resolving key safety issues. [This goal supports the NRC's Nuclear Waste Safety and Nuclear Materials Safety strategic arenas and the specific Performance Goal to make NRC activities and decisions more effective, efficient, and realistic.]**

Objective 1: *Keep abreast of cutting-edge methods and technologies being developed and utilized worldwide that are applicable for assessing and managing risks associated with cleanup, disposal, and storage of nuclear waste.*

Objective 2: *Advise the Commission on projected or perceived technical shortcomings in NRC staff capabilities that could adversely affect the agency's ability to address safety issues.*

Goal 3: **Advise the NRC on how to increase its reliance on risk as a basis for decisionmaking, including using risk assessment methods for waste management that (1) implement a risk-informed approach, (2) quantify and reveal uncertainties, and (3) are consistent across programs, where possible. [This goal supports the NRC's Nuclear Waste Safety and Nuclear Materials Safety strategic arenas and the specific Performance Goal to reduce unnecessary regulatory burden on stakeholders.]**

Objective 1: *Encourage the NRC staff and propose approaches to gain a better understanding of the inherent risks of licensed activities regarding nuclear waste disposal,* *cleanup, and materials, as well as the relationship between regulations, cost, and safety.*

Objective 2: *Encourage the NRC staff to develop an overall flexible RIPB framework for management of nuclear waste disposal, cleanup, and materials that will allow for greater transparency of the underlying assumptions and associated uncertainties of risk assessments, greater consistency across programs, and development of more defensible regulations that are linked to safety.*

Goal 4: **Support the NRC in improving public involvement in its waste and materials program and gaining increased public confidence and respect. [This goal supports the NRC's Nuclear Waste Safety and Nuclear Materials Safety strategic arenas and the specific Performance Goal to increase public confidence.]**

Objective 1: *Provide opportunities through the Federal Advisory Committee Act process for more meaningful public involvement in the regulatory process.*

Objective 2: *Recommend ways for the NRC to gain more meaningful public involvement in the regulatory process, taking into consideration international experience.*

Objective 3: *Assist the NRC in making the agency's decisionmaking process more transparent and ensuring agency documentation is thorough, clear, and readily understandable.*

Goal 5: **Improve the effectiveness and efficiency of ACNW operations. [This goal supports the NRC's Corporate Management Strategies to employ innovative and sound business practices.]**

Objective 1: *Increase the perceived value of ACNW advice to the Commission and the staff.*

Objective 2: *Improve and modify existing operational procedures to accomplish "more with less."*

PRIORITY ISSUES AND PROCESS IMPROVEMENTS

In support of its first four goals, the ACNW has identified its highest priority issues for this year, along with other important issues it plans to address this year or next, time and resources permitting. Also identified are the criteria the Committee uses to select its priority issues. In support of its fifth goal, the ACNW has identified process improvements it plans to continue to implement this year to improve its effectiveness.

The highest priority issues of 2000 are identified as first-tier priorities, and other important issues are identified as second-tier priorities. The Committee plans to conduct in-depth information gathering on most of the first-tier topics, whereas it does not plan to carry out a concentrated effort this year on most of the second-tier issues, unless directed by the Commission or in response to changes in nuclear waste legislation. The Committee may move several of these topics to the first tier in its next action plan. Each priority issue supports one or more of ACNW's goals, as indicated.

For each priority issue addressed, the Committee plans to prepare a task action plan that identifies the nature and scope of the issue and a strategy for addressing it, including planned products and schedule, and performance measures and targets that will enable the Committee to determine whether it has achieved its goals.

CRITERIA FOR SELECTING PRIORITY ISSUES

The following criteria are used to select priority issues:

- issues that are requested by the Commission or the Commissioners for ACNW review,
- the protection of public health, workers, and the environment from adverse effects of the management of nuclear waste, especially in regard to disposal facilities, that is, the risk significance of an issue,
- issues for which the ACNW's review is "self-initiated" rather than "reactive,"
- timeliness based on when an issue is scheduled to come before the Commission and when the advice would be of greatest benefit to influence the Commission's regulatory decisions,
- the relationship of an issue to the NRC's Strategic Plan, including trends and directions in regulatory practice, such as the adoption of an RIPB method of regulation and decision-making,
- issues that arise from strategies and activities of licensees and applicants,
- the potential for or likelihood of an issue to pose undue risk or costs to society, and
- issues that arise that are based on the scientific and technical information supporting the safety and performance assessments of nuclear waste disposal facilities, including the quality and level of expertise involved.

FIRST-TIER PRIORITY ISSUES

1. **Site Suitability and License Application** — The DOE is required to make a site suitability determination in 2001, and the NRC staff will comment on the sufficiency of DOE's determination in May 2001. The ACNW will begin interactions with the NRC staff on its strategy for site characterization sufficiency comments beginning in March 2000. A review plan will be developed with milestones for the Committee, the NRC staff, and DOE interactions over the next 14 months so that the ACNW will be positioned to provide advice to the Commission before the NRC's sufficiency comments are sent to the DOE.

If the Secretary of Energy recommends the Yucca Mountain Site to the President, and the President considers the site justified for application to the NRC for construction authorization, the President will submit a recommendation of the site to Congress. If there are no objections to the site from the Governor or legislature of Nevada, or if there is and Congress passes a joint resolution of repository siting approval and the President signs it into law, a license application for construction authorization would be submitted by the Secretary of Energy within 90 days. The license application would be based on a particular facility design. The ACNW will review the construction authorization request in parallel to the NRC staff's review over the 3-year statutory time period for a licensing decision. The ACNW will consider repository design and quality assurance issues under this item. This issue supports ACNW Goals 1 through 4.

2. <u>**Risk-Informed, Performance-Based Regulatory Framework**</u> — The ACNW will continue to support the agency's effort to implement a risk-informed and incrementally performance-based regulatory framework. Specifically, the ACNW and the Joint ACRS/ACNW Subcommittee will continue to encourage and help the NRC staff in developing and implementing an overall RIPB framework for nuclear waste and materials. The Committee anticipates continuing to encourage the NRC to adopt regulatory approaches that are transparent, to enhance public understanding of the key safety issues, and to encourage the NRC to use risk as a basis for setting priorities. In particular, the Committee will continue to stress the need for RIPB risk assessments to quantify the contributions of individual barriers for waste isolation and for the staff to develop guidance that clarifies its intentions regarding quantification of barriers. Issues to be addressed under this action plan item will be efforts related to clarifying the meaning of defense in depth in the nuclear waste and materials arena, the completion of the review of NRC's HLW regulation, 10 CFR Part 63, the staff's Branch Technical Position on Low-Level Waste Performance Assessment, and the continuation of the effort begun last year on risk communications. This issue supports ACNW Goals 1 through 4.

3. <u>**Decommissioning**</u> — Decommissioning topics will continue to be a primary focus of the Committee through the coming year. Areas of continued focus include guidance for implementing the final Rule on Radiological Criteria for License Termination, such as guidance on dose assessment modeling and parameter selection criteria for decommissioning assessment and streamlining the Site Decommissioning Management Program. The Committee will focus on an integrated approach to decontamination and decommissioning screening codes. RESRAD and DandD comparison reports will be reviewed along with the multi-agency decision support system used to support decommissioning decisions. Decommissioning options such as rubblization and entombment will continue to receive attention. The Committee will comment on the clearance rule. The ACNW expects to review guidance on decommissioning, such as the standard review plan for decommissioning. The Committee will take up the topic of residual contamination after decommissioning and the unrestricted versus restricted release of decommissioned sites. This issue supports ACNW Goals 1 through 4.

4. <u>**Yucca Mountain Review Plan**</u> — The ACNW will review the license application acceptance criteria as it is developed and listed in the Yucca Mountain Review Plan (YMRP). The YMRP is based on the Issue Resolution Status Reports that have documented the status of and acceptance criteria and status of each key technical issue. The Committee intends to review both pre-closure and postclosure safety issues and to ensure that the

review framework is risk informed and performance based. The ACNW will review the YMRP to ensure reviews are prioritized on the basis of risk significance. Issues such as plans for waste retrieval and pre-closure and postclosure performance confirmation and closure of individual key technical issues will be reviewed under this issue. The Committee will offer formal comments on the complete draft review plan during the public comment period beginning September 2000 and on the final review plan in September 2001. This issue supports ACNW Goals 1 through 4.

5. Transportation — The transportation of HLW and spent fuel is an issue that creates public concern. The ACNW plans to focus attention on this topic in the coming year, expanding its review of transportation issues undertaken during the review of the Yucca Mountain draft environmental impact statement. The focus will be an examination of past efforts to ensure transportation safety, such as the demonstration of cask strength and a discussion of the responsibilities of the various Government agencies involved in transportation safety. A goal would be to increase public confidence in this aspect of waste management using a risk-informed approach. The Committee also anticipates reviewing proposed changes to the NRC's transportation rule (10 CFR Part 71) from an RIPB perspective and taking a fresh look at transportation hazards through an updated modal study. This issue supports ACNW Goals 1 through 4.

SECOND-TIER PRIORITIES

1. Research — The ACNW will continue to report yearly to the Commission on NRC's waste-related research and technical assistance programs. As in past years, the ACNW will provide a chapter on waste-related research to the ACRS' annual research report to the Commission. The Committee will examine research performed by the Office of Nuclear Regulatory Research and technical assistance performed at the Center for Nuclear Waste Regulatory Analyses. The Committee expects to conduct its review of the Center's activities in San Antonio, Texas. The ACNW will continue to monitor the NRC's research program to ensure that it is changing in response to the agency's shifting emphasis to RIPB regulation. This issue supports ACNW Goals 1 through 3.

2. Low-Level Radioactive Waste and Agreement States Program — The ACNW believes that, from a risk perspective, the national low-level radioactive waste program is of growing concern because of the failure of the Low-Level Waste Policy and Amendments Act of 1986 process to bring about new LLW sites. The ACNW will consider the role of the NRC in LLW disposal from the perspective that lack of progress of the national LLW program could interfere with society's benefitting from the use of nuclear material, and therefore with NRC's ability to carry out its mission. The ACNW may examine interactions between NRC and Agreement and non-Agreement States, and whether communications can be improved. Other topics under this priority may include review of the mixed-waste (waste with a hazardous and radioactive component) issue, including the effort by the NRC and EPA to end the dual regulation of mixed wastes. The Committee will investigate LLW management practices in other countries. This issue supports ACNW Goals 1 through 4.

3. Risk Harmonization — By risk harmonization the Committee is suggesting that regulations within and across agencies reflect the same degree of protection for the relative risk

posed by similar hazards. Thus, two different agencies, both dealing with the hazard of exposure to ionizing radiation, would set like standards, or for different hazards an individual would be protected to the same degree from, for instance, the risks associated with hazardous chemicals, or ionizing radiation. The ACNW believes that adoption of an RIPB framework could advance efforts on risk harmonization and could alleviate conflicts associated with dual regulatory authority by providing a systematic and quantitative framework for assessing and comparing risk assessment approaches across and within agencies. Differences in the approach to regulating HLW, mixed wastes, and decommissioning sites between the NRC and EPA would be explored. Relative risks, such as the radiation hazard associated with the transportation of nuclear waste versus the hazard associated with traffic accidents, can also be compared. This issue supports ACNW Goals 1 through 4.

PRIORITY OPERATIONAL ACTIVITIES

Operational processes or activities that the ACNW plans to implement this year in support of ACNW Goal 5, "Enhance the Effectiveness and Efficiency of ACNW Operations," follow.

Strategic Planning — On an annual basis, the ACNW will conduct top-down planning to identify primary goals and priority issues and activities for the coming year, followed later in the year by a self-assessment of the Committee's performance against these goals. The ACNW has established performance goals and indicators to measure effectiveness and will use stakeholder surveys to solicit feedback from the public on the Committee's effectiveness.

Changes in Operational Procedures — To improve its efficiency and effectiveness, the ACNW will try to modify some of its processes and products, including the letter-writing process, the depth and consistency of advice, the scope and duration of meetings, interactions with Commissioners, communication between members and ACNW staff, and use of ACNW consultants. The Committee plans to implement the following:

- Hold more informal meetings on technical topics between individual ACNW members and members of the NRC staff.
- Allocate more time for Committee discussion of the content of letters before preparing a first draft. Circulate draft letters before the next Committee meeting so as to increase letter-writing efficiency within the bounds of the Federal Advisory Committee Act.
- Place recommendations up front, and indicate which of the recommendations the ACNW would like the NRC staff to formally respond to. If possible, the ACNW will suggest the time frame within which the staff should carry out the recommendation.
- For each priority topic, identify whether a consultant is needed and develop a list of possible consultants.
- Spend the same amount of time on Committee deliberation as is spent on the technical briefings.

- Consider reserving an entire day of every meeting for letter writing, Executive Director for Operations' response review, and discussing the Committee's future agenda.

- Conduct more meetings one-on-one with individual Commissioners and have more public interactions with the Commission.

UPDATING THIS PLAN

The ACNW will conduct periodic planning meetings to update this action plan as necessary. Revisions to the plan may be based on input from the Commission, changes to the NRC Strategic Plan, results from stakeholder surveys and self-assessments, external events and factors, and available resources.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

June 16, 2000

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: John T. Larkins, Executive Director ACRS/ACNW

SUBJECT: DRAFT REGULATORY GUIDES DG-1067, "DECOMMISSIONING OF NUCLEAR POWER REACTORS," AND DG-1071, "STANDARD FORMAT AND CONTENT FOR POST-SHUTDOWN DECOMMISSIONING ACTIVITIES REPORT"

During the 119th meeting of the Advisory Committee on Nuclear Waste, June 13-15, 2000, the Committee reviewed the subject draft regulatory guides. The Committee has no objection to the issuance of these regulatory guides.

Reference:
U.S. Nuclear Regulatory Commission "Draft Regulatory Guides DG-1067, "Decommissioning of Nuclear Power Reactors," and DG-1071, "Standard Format and Content for Post-shutdown Decommissioning Activities Report," April 24, 2000.

cc: A. Vietti-Cook, SECY
J. Craig, OEDO
G. Millman, OEDO
R. Zimmerman, NRR
M. Masnik, NRR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

June 29, 2000

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Dr. Travers:

SUBJECT: NRC EVALUATION OF DOE'S SITE RECOMMENDATION CONSIDERATIONS REPORT

During its 118th meeting, March 27–29, 2000, the Advisory Committee on Nuclear Waste (ACNW) heard presentations from the NRC staff about development of a strategy to review the Department of Energy's (DOE's) Site Recommendation Considerations Report (SRCR) and the staff's strategy to prepare sufficiency comments. The ACNW recognized that the strategy is a work in progress. The staff briefed the ACNW on the purpose, scope, objectives, integration of the strategy with ongoing activities, schedule for completion, stakeholder involvement, and proposed interactions with the ACNW.

At its 119th meeting, June 13–15, 2000, representatives from DOE briefed the ACNW on planned updates to the DOE's Repository Safety Strategy (RSS). When the revised RSS is released, it should provide important information to the NRC staff for reviewing the SRCR. The staff's approach to its sufficiency review appears to be well thought out, logical, and consistent with the overall risk-informed strategy outlined in the draft 10 CFR Part 63.

The Committee offers the following comments:

1. The ACNW plans to follow closely how the staff integrates its sufficiency review with the development and application of the Yucca Mountain Review Plan (YMRP) and with activities to resolve issues. The ACNW requests that the staff provide an example of how they plan to evaluate a specific Process Model Report (PMR) for sufficiency, using the staff's sufficiency strategy, accompanying guidance, and the YMRP.

2. The ACNW wants to gain a better understanding of how the staff prioritizes open issues, using the process for resolving key technical issues while considering an issue's importance to performance and to DOE's RSS. Specifically, the ACNW is interested in the extent to which the number and priority of open items will influence the NRC staff's sufficiency evaluation of DOE's SRCR.

3. During the NRC staff presentation to the ACNW, the concept of using conservatism as a counterbalance to uncertainty was discussed. The ACNW is skeptical about the use of

"conservatism" to compensate for uncertainty in performance analysis. Especially because overestimating consequences is not necessarily conservative. This topic may be one that the ACNW and the NRC staff should explore together so that the basis for positions on this issue are better understood.

Because of the importance of the NRC's review of the SRCR, the ACNW has drafted its own detailed plan for review of the SRCR, including a schedule to review selected supporting documents and to conduct interactions with NRC staff, DOE, and others. A copy of that draft plan is attached for your information. The ACNW looks forward to meeting with the staff throughout the review period of the SRCR.

Sincerely,

B. John Garrick
Chairman

Attachment:

"ACNW Task Action Plan for 2000 Action Plan Tier-One Priority, Site Suitability and License Application"

ADVISORY COMMITTEE ON NUCLEAR WASTE
TASK ACTION PLAN

For

2000 Action Plan Tier One Priority

SITE SUITABILITY AND LICENSE APPLICATION

Task Action Plan

Tier One Priority: Site Suitability and License Application

Lead ACNW Member: George Hornberger

Lead ACNW Staff Member: Lynn Deering

Purpose of the Plan

The purpose of this plan is to outline a strategy for the Committee to advise the Commission on the NRC staff's sufficiency review of the Department of Energy's (DOE's) Site Recommendation Considerations Report (SRCR) in Fiscal Years FY 2000 and FY 2001. This activity is a first-tier priority in ACNW's 2000 action plan (Ref. 1). A primary activity in advising the NRC on its sufficiency review will be for the Committee to conduct its own independent review of the SRCR and technical basis documents. The scope of this review will be limited by available resources, discussed further under "Review Scope and Strategy." Other major activities will include reviewing the draft Yucca Mountain Review Plan (YMRP) and the NRC staff's application of the YMRP to review the SRCR, and, finally, reviewing the staff's sufficiency comments. The Committee will interact informally with the NRC staff throughout the staff's review of the SRCR and development of the YMRP. The ACNW's approach to review the YMRP will be described in a separate task action plan.

This task action plan describes the purpose, objectives, and scope of the ACNW's review of the SRCR and technical basis documents and its desired outcome of the review; past and present related activities; planned information-gathering activities, including DOE and NRC staff briefings of the Committee and attendance at outside meetings; responsibilities of the ACNW staff, Committee members, and consultants; and products and schedules.

Purpose and Objectives of Reviews

The primary purpose of the ACNW's conduct of an independent review of the SRCR is to become familiar with DOE's approach and analysis so that the Committee can evaluate the NRC's review of the SRCR and sufficiency comments. Also, the ACNW anticipates that the Commission will request ACNW comments on the SRCR, as it did with the DOE's Viability Assessment (VA). In addition, conducting its own review of the SRCR will allow the Committee to interact with the NRC staff during its sufficiency review rather than becoming involved after the staff completes its review. Finally, the ACNW's independent review of the DOE's SRCR should enhance public and stakeholder confidence in the NRC's sufficiency review.

The ACNW's objectives in reviewing the staff's review of the SRCR and sufficiency comments include the following: (1) evaluate whether the NRC's guidance (YMRP) and approach for reviewing the SRCR reflect a risk-informed and performance-based (RIPB) approach; (2) evaluate whether the staff's sufficiency comments are logical, defensible, and focused on the most risk-significant issues; (3) identify gaps in the NRC's tools, guidance, and capability (if any) to review a license application (LA), as well as identify strengths; and (4) identify what the NRC needs to do between now and when the LA is submitted.

To position itself to conduct a review of the NRC's sufficiency review, the Committee's objectives in reviewing the SRCR and technical basis documents include evaluating (1) whether the DOE's overall approach in the Repository Safety Strategy (RSS) and the Total System Performance Assessment-Site Recommendation (TSPA-SR) is defensible; (2) whether the assumptions in the TSPA-SR are transparent, traceable, and reasonably supported on the basis of existing or planned data; (3) whether DOE's treatment of uncertainty and multiple barriers is transparent and defensible; and (4) whether the DOE has done a good job of assessing the work that it needs to do between now and submission of an LA.

Desired Outcome of Review

The ACNW's desired outcome for this Tier One priority is providing useful, high-quality advice to the Commission; bringing to the Commission's attention any vulnerabilities in the staff's capability, guidance, or other tools to review an LA for Yucca Mountain; identifying strengths; helping the Commission identify what, if anything, the staff needs to do between now and submittal of an LA; and evaluating whether the staff has a logical, defensible, RIPB basis for its findings that are obviously linked to safety. Overall, the ACNW would like its advice on the SRCR and the staff's sufficiency review to help the Commission make better, informed decisions about Yucca Mountain site sufficiency with a high degree of public confidence.

Background and Rationale for Review

The Committee has identified the SRCR as a first-tier priority in its 2000 action plan because the NRC's review of the DOE's SRCR and preparation of sufficiency comments is a high-priority activity of the Commission having national significance. The Nuclear Waste Policy Act (NWPA) requires DOE to make a site suitability recommendation to the President, which is currently planned for May 2001. The NWPA requires that the DOE's site suitability recommendation include preliminary comments from the NRC concerning the extent to which the at-depth site characterization analysis and waste form proposal seem to be sufficient for inclusion in an LA.

The NRC's review has national significance for at least two reasons. First, the NRC staff's sufficiency review will serve as an indicator of whether the staff has the tools, guidance, and capability to review an LA for Yucca Mountain, including whether there are gaps in NRC's existing program and what, if anything, the agency must do to position itself to review an LA. Second, the NRC's review will indicate, from the regulator's perspective, whether DOE has enough data and conceptual understanding of the system to develop a safety case for the LA. NRC's review will have implications regarding whether DOE decides to recommend the Yucca Mountain site and eventually submit an LA.

The DOE's current schedule calls for release of the SRCR in mid-December 2000 and for the NRC to provide its comments by May 25, 2001. The NRC staff will provide its strategy for conducting its sufficiency review to the Commission by June 30, 2000. The staff is developing the YMRP guidance in parallel with the strategy so that the staff can use the YMRP in reviewing the process model reports (PMRs) and the SRCR. The staff currently plans to release Revision 1 of the YMRP in September 2000 and to brief the ACNW at that time, in advance of DOE's formal request to NRC to provide its sufficiency comments. The ACNW was briefed on the draft strategy for sufficiency in March 2000 and will begin informal interactions with the NRC staff on its strategy for site characterization sufficiency comments and Revision 1 of the YMRP beginning in June 2000.

The purpose of the staff's review is to evaluate whether DOE has enough data and conceptual understanding of the system to develop a safety case for the LA. The staff's documented review will serve as a progress report on DOE's sufficiency of data, design, analyses, and plans for the LA, and on the status of the Key Technical Issues (KTI) issue resolution. The NRC staff will evaluate sufficiency in the context of the NRC's performance-based approach to licensing as proposed in draft 10 CFR Part 63. The review is to be fully integrated into the NRC's licensing strategy outlined in the YMRP and the KTI issue resolution process. The staff will not remark on DOE's dose estimate, nor will it review the document against DOE's proposed siting guidelines in 10 CFR Part 963.

Past Related ACNW Activities

In 1999, the ACNW conducted an independent review of the DOE's VA at the request of the Commission. The Commission also requested the Committee to review and comment on the draft High-Level Waste (HLW) rule for Yucca Mountain, 10 CFR Part 63. The Committee provided comments on the proposed final 10 CFR Part 63 in early 2000. Over the past several years, the Committee has offered advice on implementation of RIPB regulation, including the agency RIPB white paper, implementation of defense-in-depth and multiple barriers concept in draft 10 CFR Part 63, transparency in performance assessment, and implementing a risk-informed framework for NMSS. Other related activities include developing white papers on DOE's design selection process and repository design, and on the repository near-field chemistry, and a letter on the Engineered Barrier System (EBS) for the proposed repository.

Review Scope and Strategy

<u>Scope</u>

Once the DOE issues the SRCR in December 2000, the ACNW will review relevant portions of the SRCR, as well as the NRC staff's sufficiency comments, currently scheduled for completion in May 2001. Before DOE submits the SRCR, the ACNW will review, to the extent possible, Revision 4 of the RSS, the SR-design, the TSPA-SR, selected PMRs and analysis and model reports (AMRs), selected issue resolution status reports (IRSRs), the TSPA methods and assumptions document, and possibly other technical basis documents as identified. Further, the Commission has requested that the ACNW review and comment on Revision 1 of the YMRP. This document will be presented to the ACNW in September 2000, and the Commission expects the ACNW to help the staff in developing the document before the September 2000 meeting. The YMRP is a closely related activity because the staff plans to use Revision 1 of the YMRP to conduct its sufficiency review. Other related activities include evaluating the staff's overall capability to review a license application, including the TPA Code 4.0, and the Code peer review report, which may be explored as part of the ACNW's annual research review at the Center for Nuclear Waste Regulatory Analyses (CNWRA) in November 2000. Because of time and resource constraints, the Committee expects to review many of these documents informally rather than conducting a formal review and providing comments to the Commission. Selected PMRs, AMRs, and IRSRs may fit in this category.

Strategy

The ACNW members, staff, and consultants will be assigned a lead role on various documents or portions of documents corresponding to their areas of expertise. Individuals will form into teams to conduct the reviews. Each team will be responsible for coordinating with the NRC staff on their areas of responsibility, as well as interacting with and evaluating their consultants' input. George Hornberger of the ACNW and Lynn Deering of the ACNW staff will be responsible for consolidating all of the review comments and orchestrating development of the draft letter on the SRCR to be issued by May 2001. General areas of responsibility and assigned PMRs are identified in Tables 1 and 2, respectively. Obviously, because of resource and time limitations, the Committee cannot review all portions of all documents. High priority will be given to the PMRs that correspond to one or more of DOE's principal factors,[1] including (1) unsaturated zone flow and transport, (2) saturated zone flow and transport, (3) waste package degradation, (4) waste form degradation, (5) biosphere, and (6) disruptive events. High priority will also be given to reviewing the TSPA methods and assumptions document. A medium priority rating will be given to all other PMRs, including near-field environment and EBS degradation. Finally, a low priority rating will be assigned to reviewing the integrated site model PMR.

To further focus the ACNW's review, a review plan or a template providing guidance to the reviewers is being developed for reviewing the RSS, the PMRs, and the AMRs, the YMRP, the TSPA-SR, and eventually, the SRCR. The templates should consist of several or more key questions that will guide and focus the Committee's review and help position the Committee in preparing letter reports on the SRCR, the YMRP, and other reviews.

Planned information-Gathering Activities

Table 3 contains a list of briefings of the ACNW related to the Committee's review of the SRCR and related documents, the YMRP, and the staff's sufficiency review. Table 4 lists NRC-DOE Technical Exchanges and Appendix 7 meetings and outside meetings that the Committee members, staff, or consultants plan to attend.

ACNW-NRC Staff Interactions

Table 5 lists planned informal interactions between individual Committee members and NRC staff on selected technical topics.

Schedule

See Figure 1, attached.

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[1]The DOE's principal factors as of an M&O briefing of the Committee on June 14, 2000, include seepage into drifts, drip shield performance, waste package performance, dissolved radionuclide concentrations, colloid-associated radionuclide concentrations, unsaturated zone radionuclide transport, saturated zone radionuclide transport, biosphere dose conversion factors, igneous activity probability, and igneous activity repository effects.

Updating this Plan

This task action plan will be updated monthly and included in the Committee's meeting notebook.

Reference

Letter from John Garrick, Chairman, ACNW, to Chairman Meserve, "ACNW Action Plan and Priority Issues," April 18, 2000.

TABLE 1 – ASSIGNED AREAS OF RESPONSIBILITIES

TEAM MEMBER	GENERAL AREAS OF RESPONSIBILITY
George Hornberger , ACNW	Lead on SR review, saturated and unsaturated zone flow & transport, disruptive events, coupled processes, radionuclide transport, natural analogs, multiple barriers, KTI issue resolution
Raymond Wymer, ACNW	Waste form and waste package degradation, radionuclide transport, near-field environment, corrosion, natural analogs, multiple barriers, coupled processes
John Garrick, ACNW	YMRP, integrated safety assessment (ISA), TSPA-SR, DID, multiple barriers, RIPB, FEPS, biosphere
Milton Levenson, ACNW	Repository design, ISA, EBS degradation, thermal loads, coupled processes, performance confirmation
Lynn Deering, ACNW staff	Staff lead on SR review; sat and unsaturated zone flow and transport, natural analogs, disruptive events, KTI issue resolution
Amarjit Singh, ACNW staff	Preclosure issues, (ISA), waste form degradation, waste package design
Richard Savio, ACNW staff	performance confirmation, waste package design, corrosion, EBS degradation
John Larkins, ACNW Director	Oversight of SR review, performance confirmation
Howard Larson, ACNW staff	KTI issue resolution, RES
Richard Major, ACNW staff	YMRP, ISA, repository design, thermal load, coupled processes
Andrew Campbell, ACNW staff	EBS, waste form and waste package degradation, radionuclide transport, TSPA-SR, FEPs, near-field environment, natural analogs

TABLE 2 — PMR ASSIGNMENTS

PF= principal factor based on June 14, 2000 DOE briefing on RSS
OF= other factors noted in table 3-3 of Rev 3 of DOE's Repository Safety Strategy (Ref. 2)

PMR	ACNW PRIORITY	PRINCIPAL FACTORS	KTI	TEAM MEMBERS AND STATUS OF PMR
UZ F&T	High	PFs =Seepage, retardation in UZ. PMR describes processes affecting amount of water entering UZ above repository that could contact waste and the movement of water thru the UZ below the repository and potential transport of radio- nuclides in that water.	UZ and SZ Flow; Rad Transport	Hornberger, Deering, TBD * draft PMR has been provided to the NRC staff, waiting for final PMR to begin review
SZ F&T	High	PFs= retardation in SZ; dissolved radionuclide concentrations. PMR describes processes that control the movement of water thru the sat zone below the repository and the distribution of dissolved rads or colloids that might be released and migrate to the sat zone, and dilution of rad concentrations during migration thru sat zone.	UZ and SZ Flow; Rad Transport	Hornberger, Deering,TBD *draft not yet provided

TABLE 2 — PMR ASSIGNMENTS (CONT'D)

PMR	ACNW PRIORITY	PRINCIPAL FACTORS	KTI	TEAM MEMBERS AND STATUS OF PMR
Waste Package Degradation	High	PFs= performance of drip shield; performance of waste package barriers. PMR describes processes that could lead to drip shield and waste package degradation e.g. corrosion of waste package materials in the near-field environment.	Container life	Wymer, Major, TBD * draft PMR has been provided to the NRC staff, waiting for final PMR to begin review
TSPA-SR Methods and Assumptions	High	All	Total System Performance Assessment and Integration	Garrick, Campbell,TBD * draft PMR has been provided to the NRC staff, waiting for final PMR to begin review
Waste Form Degradation	High	PFs= colloid associated radionuclide concentrations. PMR describes waste characteristics that limit the rate of release of rads. Processes include waste canister degradation, cladding degradation, and waste form dissolution. Describes the manner in which waste forms degrade and expected rad releases.	Container Life Source Term Rad Transport	Wymer, Campbell, TBD * draft PMR has been provided to the NRC staff, waiting for final PMR to begin review

TABLE 2 — PMR ASSIGNMENTS (CONT'D)

PMR	ACNW PRIORITY	PRINCIPAL FACTORS	KTI	TEAM MEMBERS AND STATUS OF PMR
Biosphere	High	PFs= Biosphere dose conversion factors. PMR describes characteristics of biosphere that influence transport of rads to humans.	Total System Performance Assessment and Integration	Garrick, Major, Kearfott * Draft PMR has been provided to the NRC staff, waiting for final PMR to begin review
EBS Degradation F&T	Med	OFs = Environments on drip shield; Transport through drift invert. PMR describes processes that would lead to degradation of the EBS and affect movement of rads thru those barriers Provides info about thermal, hydro, and geochemical processes acting on engineered barriers.	Evolution of Near Field; Thermal effects on Flow	Levenson, Major, TBD * draft PMR has been provided to the NRC staff, waiting for final PMR to begin review
Disruptive events	High	PMR describes tectonic properties that could disrupt repository system	Structural Deformation and Seismicity, Igneous Activity	Hornberger, Deering, Hinze *waiting for final PMR to begin review

TABLE 2 — PMR ASSIGNMENTS (CONT'D)

PMR	ACNW PRIORITY	PRINCIPAL FACTORS	KTI	TEAM MEMBERS AND STATUS OF PMR
Near Field Environment	Medium	OFs = Coupled process effects on seepage. PMR describes processes important to limiting the amount of water that can contact waste, including effects of heat on UZ flow at drift wall; seepage; temperature and humidity on the EBS, and chemical reactions and products and mechanical interactions in near field host rock and drifts.	Evolution of Near-field environment; repository design and thermo-mechanical effects	Wymer, Campbell, possibly Hornberger TBD *draft not yet provided
Integrated Site Model	Low	PMR describes framework for geologic properties	Structural Deformation and Seismicity	Hornberger, Deering, Hinze *final PMR has been provided to the ACNW

TABLE 3 — BRIEFINGS TO THE COMMITTEE

ACNW MEETING	SCHEDULED OR PLANNED BRIEFING	LEAD MEMBER AND STAFF	CONSULTANT * = INVOLVED BUT NOT PRESENT AT MEETING
June 13- 15, 2000	1. DOE Briefing on RSS	1. Hornberger, Deering	None
	2. DOE Briefing on SR-Design	2. Levenson, Major	None
	3. DOE Briefing on 963	3. Singh, Hornberger	None
	4. ACNW Staff Briefing on Proposed Strategy for ACNW 's review of SR	4. Hornberger, Deering	None
	➘ <u>Product</u> - Task Action Plan	completed	
	➘ <u>Product</u> - Letter Report To EDO on Staff's Sufficiency Strategy	completed	
	5. Informal Discussions with NRC Staff on YMRP	5. Garrick, Major	None

TABLE 3 — BRIEFINGS TO THE COMMITTEE (CONT'D)

ACNW MEETING	SCHEDULED OR PLANNED BRIEFING	LEAD MEMBER AND STAFF	CONSULTANT * = INVOLVED BUT NOT PRESENT AT MEETING
JULY 25-27, 2000	1. NRC staff Briefing on Highlights of KTI Issue Resolution Technical Exchange ➘ Product - Possible Letter Report to Com-Mission on KTI Issue Resolution	1. Hornberger, Deering	
	2. June 21st, DOE Brief-ing on Performance Con-firmation ➘ Product - Possible Letter Report to Com-Mission on Perform-ance confirmation	2. Levenson, Larkins, Savio	
	3. Informal discussions with NRC staff on YMRP	3. Garrick, Hornberger, Levenson, Wymer	3. None

TABLE 3 — BRIEFINGS TO THE COMMITTEE (CONT'D)

ACNW MEETING	SCHEDULED OR PLANNED BRIEFING	LEAD MEMBER AND STAFF	CONSULTANT * = INVOLVED BUT NOT PRESENT AT MEETING
September 19-21, 2000 Las Vegas, NV	1. DOE Briefing on Final SR-Design 2. DOE Briefing on TSPA-SR 3. Working Group on YMRP Product - ACNW Letter Report to Commission on YMRP	1. Levenson, Major 2. Garrick, Campbell 3. Garrick, Major	1. None 2. TBD 3.TBD **Kick-off Meeting with Consultants Ewing, Shewmon, Clark, Kearfott
October, 17-19, 2000	TBD		
November 15-17, 2000	1. Visit to CNWRA to evaluate research, TPA Code, and technical capability	1. Hornberger, Larson	1.TBD

TABLE 4 — UPCOMING MEETINGS OF INTEREST TO ACNW

MEETING	DATE	LOCATION	MEMBER/STAFF COVERAGE
NWTRB Summer Meeting - TSPA-SR	8/1 - 8/2	Carson City, NV	Campbell
Appendix 7 UZ Flow and Transport PMR	8/16-17/00	Berkeley, CA	Davis
Biosphere PMR Technical Exchange	8/29/00 **POST- PONED**	Las Vegas, NV	Kearfott
Disruptive Events PMR Igneous Activity	8/30 - 8/31	Las Vegas, NV	Hinze
Waste Package and Waste Form Degradation PMRs Technical Exchange	9/12- 9/13/00	Las Vegas, NV	Singh
Surface and Subsurface Design SDD	10/24/00	Las Vegas, NV	Levenson/Major
Saturated Zone F&T Technical Exchange	11/1 - 2/00	Albuquerque, NM	Davis
Repository Safety Strategy & KTI Resolution	11/7 - 11/8	Berkeley, CA	TBD
BRWM - Site Recommendation	12/14 - 12/15	Wash. DC	TBD
EBS Degradation, F&T PMR; Near-field PMR	1/9 -10/01	Las Vegas, NV	TBD, Levenson

TABLE 5 — ACNW & ACNW STAFF PLANNED INTERACTIONS

ACNW MEETING	INDIVIDUALS AND SUBJECT
June 13- 15, 2000 lunch time meetings	YMRP, John Garrick, George Hornberger, R. Major w/ Jeff Ciacco, DWM
July 28-30, 2000	1. YMRP Rev. 1, John Garrick, George Hornberger, R. Major with Jeff Ciacco, DWM July 25, 3:00 - 5:00 pm 2. Sufficiency review and review of PMRs George Hornberger, L. Deering, James Firth July 26, lunch time meeting
September 2000	TBD
October, 2000	TBD
November, 2000	TBD

TABLE 6 — COMMITTEE PRODUCTS

LETTER * = Commission request	TARGET COMPLETION DATE	LEAD ACNW MEMBER ACNW AND STAFF
Letter to EDO on Staff's Sufficiency Strategy	6/00-completed	Hornberger/Deering
Possible Letter on KTI Issue Resolution	9/00	Garrick/Deering
Possible letter on performance confirmation	9/00	Levenson/Larkins
*Letter on Rev. 1 YMRP	10/00	Garrick/ Major
Staff's application of sufficiency strategy and YMRP to evaluate a PMR	11/00	Hornberger/ Deering
Letter to Commission on SRCR	4/01	Hornberger/ Deering

EXAMPLE SCHEDULE FOR ACNW REVIEW OF PMRs AND TECH BASIS DOCUMENTS

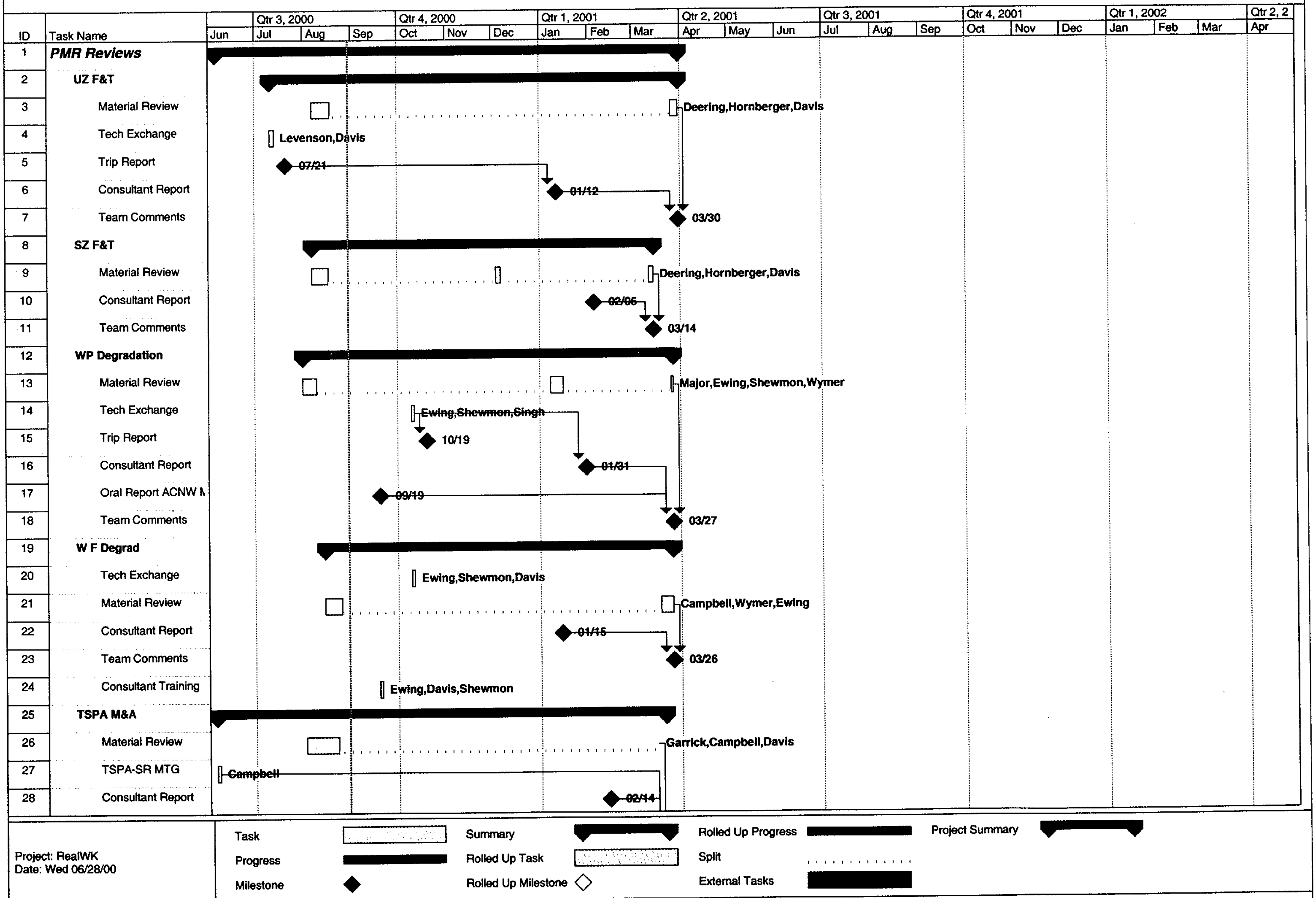
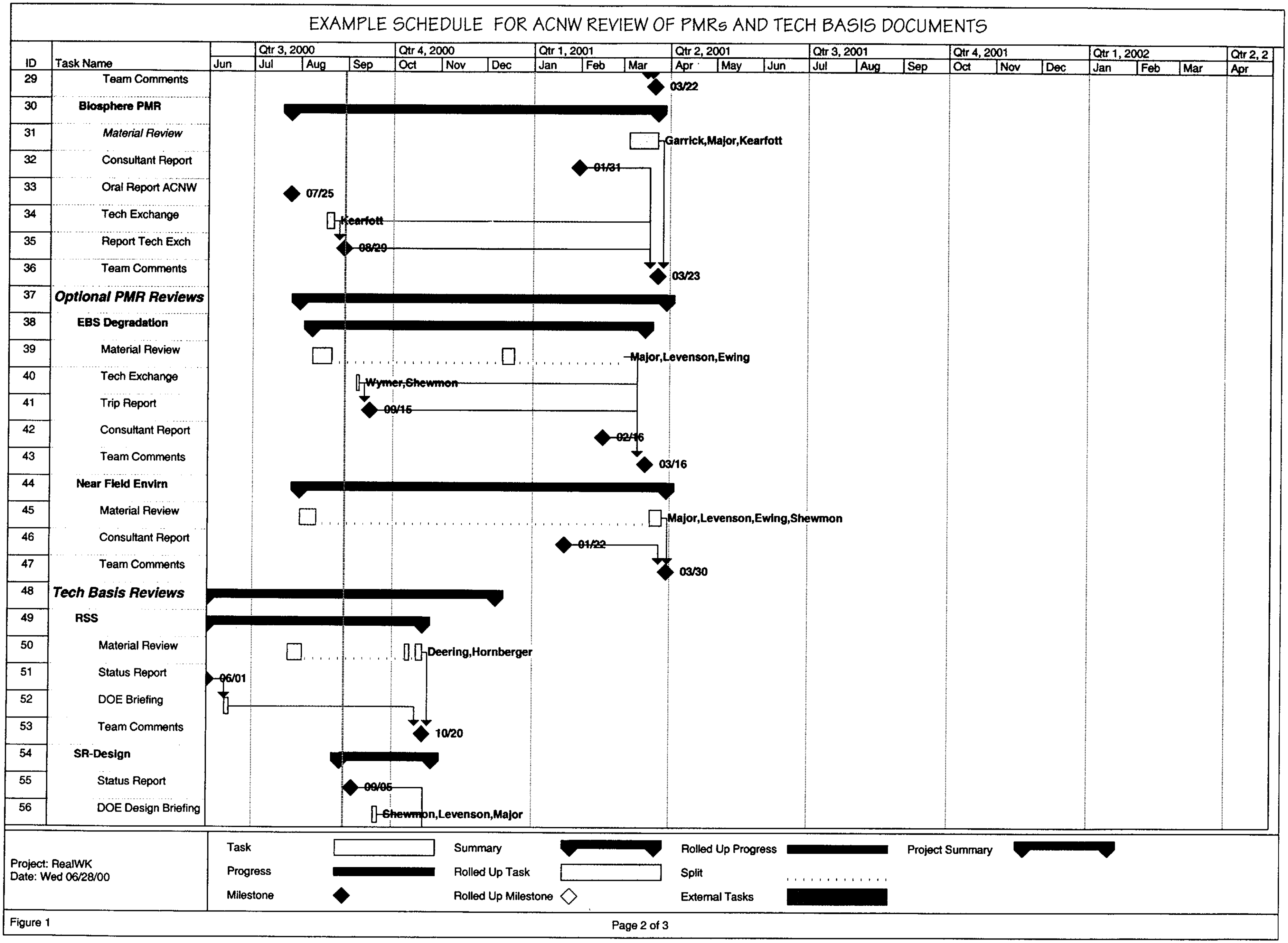


Figure 1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Task Name
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Qtr 4, 2000
Qtr 1, 2001
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Qtr 3, 2001
Qtr 4, 2001
Qtr 1, 2002
Qtr 2, 2
Jun Jul Aug Sep Oct Nov Dec Jan Feb Mar Apr May Jun Jul Aug Sep Oct Nov Dec Jan Feb Mar Apr
29 Team Comments
03/22
30 Biosphere PMR
31 Material Review
Garrick,Major,Kearfott
32 Consultant Report
01/31
33 Oral Report ACNW
07/25
34 Tech Exchange
Kearfott
35 Report Tech Exch
08/29
36 Team Comments
03/23
37 Optional PMR Reviews
38 EBS Degradation
39 Material Review
Major,Levenson,Ewing
40 Tech Exchange
Wymer,Shewmon
41 Trip Report
09/15
42 Consultant Report
02/16
43 Team Comments
03/16
44 Near Field Envirn
45 Material Review
Major,Levenson,Ewing,Shewmon
46 Consultant Report
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47 Team Comments
03/30
48 Tech Basis Reviews
49 RSS
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Deering,Hornberger
51 Status Report
06/01
52 DOE Briefing
53 Team Comments
10/20
54 SR-Design
55 Status Report
09/05
56 DOE Design Briefing
Shewmon,Levenson,Major
Project: RealWK
Date: Wed 06/28/00
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Rolled Up Task
Rolled Up Milestone
Rolled Up Progress
Split
External Tasks
Project Summary
Figure 1
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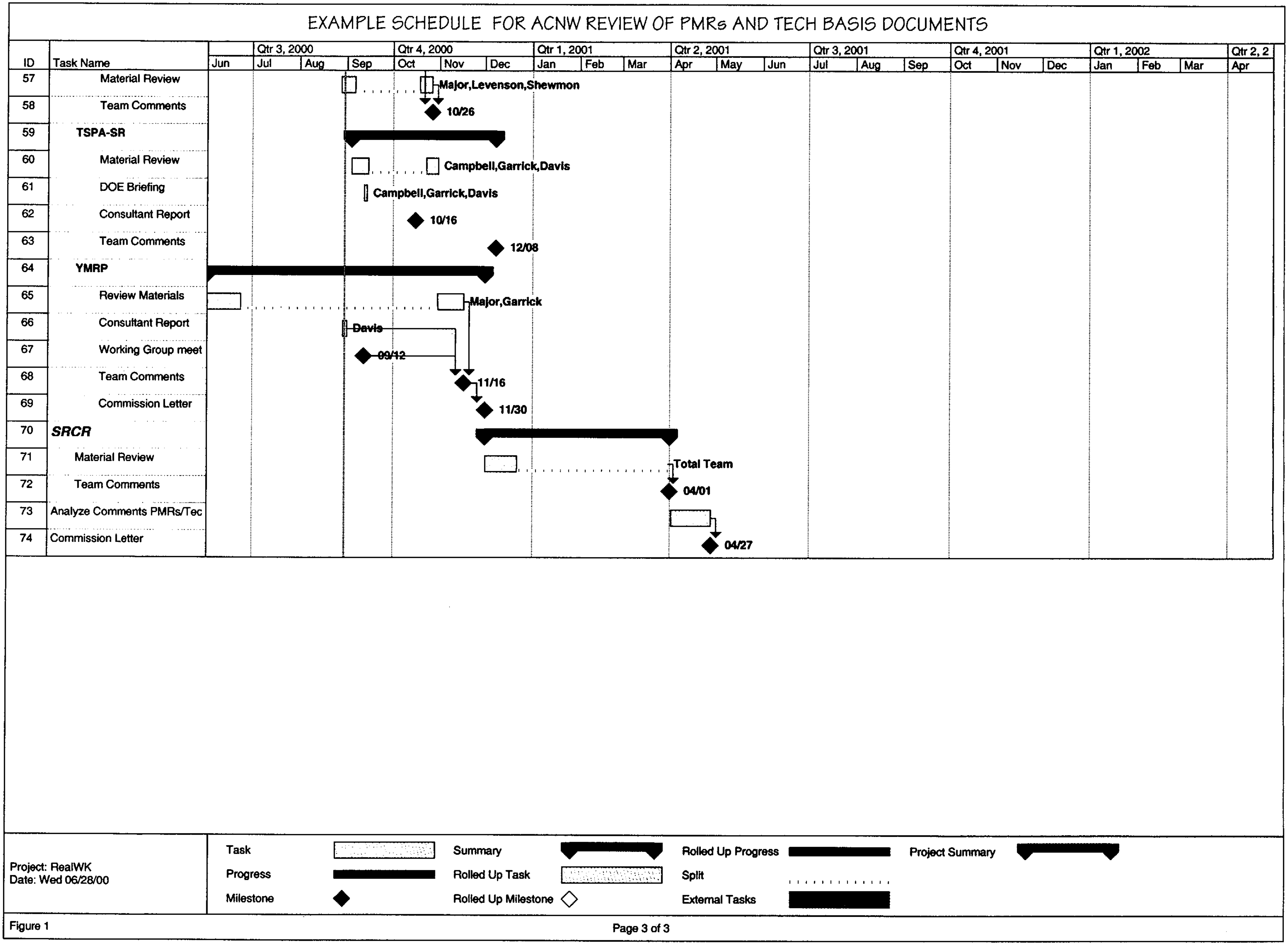
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Qtr 3, 2001
Qtr 4, 2001
Qtr 1, 2002
Qtr 2, 2
Jun Jul Aug Sep Oct Nov Dec Jan Feb Mar Apr May Jun Jul Aug Sep Oct Nov Dec Jan Feb Mar Apr
57 Material Review
Major,Levenson,Shewmon
58 Team Comments
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59 TSPA-SR
60 Material Review
Campbell,Garrick,Davis
61 DOE Briefing
Campbell,Garrick,Davis
62 Consultant Report
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72 Team Comments
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73 Analyze Comments PMRs/Tec
74 Commission Letter
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Project: RealWK
Date: Wed 06/28/00
Task
Progress
Milestone
Summary
Rolled Up Task
Rolled Up Milestone
Rolled Up Progress
Split
External Tasks
Project Summary
Figure 1
Page 3 of 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON NUCLEAR WASTE
WASHINGTON, D.C. 20555-0001

July 27, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: DEVELOPMENT OF RISK-INFORMED REGULATION IN THE OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS

Dear Chairman Meserve:

The Joint Subcommittee of the Advisory Committee on Reactor Safeguards (ACRS) and the Advisory Committee on Nuclear Waste (ACNW) met with representatives of the NRC staff on May 4, 2000, to discuss development of risk-informed regulation in the Office of Nuclear Material Safety and Safeguards (NMSS). The staff presented its activities and proposed actions to address the issues noted in the ACRS/ACNW joint report dated November 17, 1999, concerning the risk-informed framework in SECY-99-100 [References 1 and 2]. This matter was subsequently discussed at the 120th meeting of the ACNW on July 25-27, 2000. Although the ACNW has the responsibility for providing advice to the Commission on this subject, it should be noted that ACRS members Drs. Apostolakis and Kress (members of the Joint Subcommittee) participated in writing this report.

Specific topics addressed by the Subcommittee at the meeting on May 4 included risk-informed fuel cycle programs, integrated safety assessments (ISAs), byproduct material risk analysis, dry cask storage risk analysis, results of a public workshop on the use of risk information in regulating the use of nuclear materials, and related matters. Some of these issues relate to proposed modifications to 10 CFR Part 70 [Reference 3]. The ACNW and the Joint Subcommittee members from the ACRS have no objections to the proposed modifications to 10 CFR Part 70. We intend to interact with the staff to address our concerns regarding implementation of the revised rule.

The Subcommittee was impressed with the work being performed by NMSS in addressing the challenges in developing a risk-informed regulatory process. We were pleased to see progress in many areas, including the application of ISA to fuel cycle facilities, the completion of the byproduct material risk analysis, the beginning of probabilistic risk assessment (PRA) work on dry cask storage, and the transportation package performance risk studies. We were encouraged by the recent interactions between the NMSS staff and stakeholders on risk issues and believe that public workshops such as the one held in April of this year contribute to the information base needed to implement effective risk-management practices. These types of activities are important to obtain stakeholder input and to assure stakeholders that their concerns are being properly addressed.

Discussion and Recommendations

A general observation of the Subcommittee was that while several risk-assessment activities are underway in NMSS, a strong commitment and a clear vision are necessary regarding the direction to be taken in future implementation of risk-informed and performance-based regulatory practices. We recognize that there is a great diversity of nuclear material activities being regulated, covering such broad areas as fuel cycle facilities, byproduct materials, fuel storage, and transportation. Although, in general, the implementation of risk-informed regulatory practices needs to be different for different activities, it also needs to be guided by a stated overall policy or mission statement and an articulation of a set of fundamental principles.

Recommendation

- NMSS should establish a stated overall policy or a clear mission statement with supporting principles for the implementation of risk-informed and performance-based practices. The principles adopted should be consistent with the high-level principles used in reactor safety applications [Reference 4].

The proposed rulemaking amending 10 CFR Part 70 with emphasis on ISA does facilitate increased use of risk assessment in the licensing activities associated with special nuclear materials. We have some concern with the lack of visibility in the guidance documents as to the use of the agency's expertise and experience in risk assessment. We believe that the implementation guidance would benefit from input from the NRC staff with experience in formulating risk-acceptance criteria.

Recommendation

- NMSS should seek assistance from NRR and RES staffs with experience in the implementation of risk-assessment practices and the development of risk-acceptance criteria while revising the Standard Review Plan. These interactions should be conducted in a manner that solicits external stakeholder participation and feedback.

ISA has its roots in chemical safety analysis, not in nuclear safety analysis. The bulk of experience in the nuclear field is with PRA. The principal differences between ISA and PRA are in the methods of analysis and in the language and terminology employed. By taking advantage of the agency's experience with the use of PRA, NMSS could expedite the task of risk-informing the licensing activities of special nuclear materials. Inefficiencies stemming from starting anew with ISA could be avoided. The use of common terminology would contribute to the understanding and the coherent application of the risk-informed regulatory philosophy to all of the agency's activities. The Commission's White Paper on the use of PRA provides clear definitions, and there is no need to introduce different terminology [Reference 5]. The use of ISA, as compared to the use of PRA, is new to the nuclear industry. We doubt that ISA has been sufficiently tested on issues critical to nuclear regulation, such as applicable standards, peer review, quality control, ownership of analysis, validity of databases, and completeness of scope.

Recommendation

- The NMSS staff should take full advantage of the agency's experience in PRA in its application of ISA to ensure ISA's evolution to risk-informed practices of safety analysis. A common language based on the Commission's White Paper should be adopted.

The study, "Risk Analysis and Evaluation of Regulatory Options for Nuclear Byproduct Material Systems," is a significant step in answering the earlier questions that were raised by the Joint Subcommittee about how NMSS applies its own expertise to rank various risks [Reference 6]. Although the study was limited to byproduct material and did not consider a number of important issues, such as uncertainty, defense in depth, and, in the case of medical applications, patient risk, it does represent an important start to prioritizing risks.

We consider the application of PRA methods to dry cask storage to be reasonably straightforward. The study has not advanced to the point where there are specific questions about how the PRA methods were applied.

Recommendation

- A risk-informed approach of prioritizing contributors to risk should be applied to the other nuclear material categories within NMSS' area of regulatory responsibility.

Future Meetings

We look forward to meeting with the staff to discuss NMSS' vision on the underlying principles and motivation for pursuing the ISA approach. We believe that the best and most efficient way to continue discussions with NMSS on the use of ISA would be to discuss an important application. We are also interested in following the progress being made on decisions that will lead to safety goals of the different categories of nuclear material activities. Of particular interest on safety goals is the selection of risk measures. As indicated above, the Subcommittee is interested in following the development of the Standard Review Plan for special nuclear materials. Finally, because actuarial data can contribute to a risk-informed understanding of byproduct materials and devices, we are interested in learning more about the practices for recording and archiving data, for example, how the Nuclear Material Events Database is operated.

Sincerely,

B. John Garrick
Chairman, ACNW

References:
1. Report dated November 17, 1999, from B. John Garrick, Chairman, ACNW, and Dana A. Powers, Chairman, ACRS, to Richard A. Meserve, Chairman, NRC, Subject: Implementing a Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards.
2. SECY-99-100, Memorandum dated March 31, 1999, from William D. Travers, Executive Director for Operations, NRC, for the Commissioners, Subject: Framework for Risk-Informed Regulation in the Office of Nuclear Material Safety and Safeguards.
3. SECY-99-147, Memorandum dated June 2, 1999, from William D. Travers, Executive Director for Operations, NRC, to The Commissioners, Subject: Proposed Rulemaking - Domestic Licensing of Special Nuclear Material.
4. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.
5. Memorandum dated February 24, 1999, from Annette Vietti-Cook, Secretary, NRC, to William D. Travers, Executive Director for Operations, NRC, Subject: Staff Requirements - SECY-99-144 - White Paper on Risk-Informed and Performance-Based Regulation.
6. U.S. Nuclear Regulatory Commission, NUREG/CR-6642, Vol. 1-3, "Risk Analysis and Evaluation of Regulatory Options for Nuclear Byproduct Material Systems."

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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(See instructions on the reverse)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This compilation contains 11 reports issued by the Advisory Committee on Nuclear Waste (ACNW) during the Twelfth year of its operation. The reports were submitted to the Chairman and Commissioners of the U. S. Nuclear Regulatory Commission (NRC). All reports prepared by the Committee have been made available to the public through the NRC Public Document Room, or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS) which is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room); the U. S. Library of Congress, and the Committee's Web site at http://www.nrc.gov/ACRSACNW.

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