FIRST INTERIM REPORT

TOTAL SYSTEMS PERFORMANCE ASSESSMENT
PEER REVIEW PANEL

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Peer Review of the
Total System Performance Assessment-Viability Assessment

First Interim Report

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PREFACE

The Peer Review Panel has written this report with two distinct audiences in mind: (1) those who are actively engaged in producing the Total Systems Performance Assessment and (2) those who are interested in the progress of the Total Systems Performance Assessment and its implications for future policy decisions. The first group will find some of the introductory information unnecessary and will want to concentrate on the technical findings and explanations. The Panel hopes that the report also contains enough background information and explanations of terms and is written clearly enough that it will be intelligible to the second group, namely those interested in the outcome, but not involved in the technical work. The Panel welcomes comments from readers on the usefulness of this approach and how future reports can be improved.

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Executive Summary

Predicting the performance of a repository at Yucca Mountain is a technically challenging task. As is noted in Section II, the analysis involves the modeling of complex hydrologic, thermal, chemical, and mechanical interactions over times spanning tens to hundreds of thousands of years. Many of the technical questions that are necessarily addressed in this analysis, for example, the distribution of the time for corrosion to breach the waste canisters, are at the forefront of the capabilities of present day science. With these complexities in mind, the TSPA Peer Review Panel has evaluated work that has been done and that is in progress.

This first interim report reflects the Panel's initial review, subject to the following significant limitations:

- The Total System Performance Assessment (TSPA) supporting the Viability Assessment (VA) has not yet been written and, thus, the Panel is reviewing a work in progress. The Panel has available to it previous TSPA reports and various technical reports prepared in support of TSPA. The Panel also has been given briefings on the work being done for the TSPA-VA and has sent Panel members to “abstraction” meetings conducted by the TSPA-VA team to assign priorities to key technical issues and provide simplified versions of analyses suitable for an integrated analysis of the site as a whole. Members of the Panel have also attended other project workshops.

- The design of the repository is evolving. A number of aspects of the design have changed since the Panel’s first meeting or are under review. Examples include the use of Alloy 625 instead of Alloy 825 as the inner corrosion resistant material of the waste canisters, the potential use of a gravel or sand backfill in the drifts, and consideration of alternative configurations for spent fuel containing highly enriched uranium.

- The Panel is still in an orientation phase. This is a large and technically diverse project with a long history. In the few months since the Panel’s first meeting, members have not had sufficient time to review all relevant documents or to learn of all the work that is being done to support the TSPA-VA.

As is noted, some issues such as seismic contributions to risk, gaseous pathways of exposure, the performance of spent fuel cladding, and the effects of volcanism and human intrusion, are not addressed in this first interim report. Other issues such as the potential for movement of plutonium via colloids, are noted, but the Panel has not yet reached closure regarding the adequacy of the manner in which these issues are being addressed.

The Panel's initial findings are not easily condensed into a few sentences, and, for that reason, are not listed in this Executive Summary. The reader can find the major initial findings summarized in Section III. Support for the findings are provided in Sections I and II.

In Section I the Panel provides an overview of the TSPA-VA approach and constraints, and includes the Panel's understanding of (1) the project's use of both detailed deterministic models and simplified abstraction models suitable for use in an integrated probabilistic analysis, (2) the repository and the Panel's understanding of how it is intended to isolate wastes, and (3) the approach taken by the project to assess performance in the absence of applicable standards from the U. S. Environmental Protection Agency and accompanying regulations from the U. S. Nuclear Regulatory Commission.

In Section II the Panel discusses in more detail its understanding of processes and events that would affect the future performance of a repository at Yucca Mountain. This Section provides a basis and context for the Panel's initial findings and a description of the Panel's understanding of the key analytical issues associated with assessing the repository performance assessment. Section II contains the major elements listed in the Table of Contents: (1) site initial conditions; (2) site conditions as affected by the repository; (3) the engineered barrier system and isolation provided by the waste form; (4) disruptive events and criticality; (5) transport of radionuclides from the repository; and (6) the biosphere, doses, and health risks.

As noted above, Section III summarizes the major initial findings of Sections I and II.

Appendix A contains a list of questions that the Panel has identified as important to an understanding of repository performance. The questions are provided as an indication of issues the Panel plans to explore in its next phase.

An important goal in providing the Panel's understanding of the repository design and how it would perform is to identify and correct any misconceptions that exist. The Panel welcomes comments and suggestions regarding this initial report and the Panel's continuing review.

I. INTRODUCTION

In the Energy and Water Appropriations Act for fiscal year 1997, Congress specified four components of a viability assessment for a potential high level radioactive waste repository at Yucca Mountain, Nevada. These include:

> ...a total system performance assessment, based upon the design concept and the scientific data and analysis available by September 30, 1998, describing the probable behavior of the repository in the Yucca Mountain geological setting relative to the overall system performance standards.

The objective of the Total System Performance Assessment Peer Review is to provide a formal, independent evaluation and critique of the Total System Performance Assessment (TSPA) supporting the Viability Assessment (VA) for the Civilian Radioactive Waste Management System Management and Operating contractor (CRWMS M&O). The TSPA-VA is being conducted by the CRWMS M&O for the U. S. Department of Energy (DOE) Yucca Mountain Site Characterization Office. The Peer Review Panel has been asked to conduct a phased review over a two-year period during the development and completion of the TSPA-VA.

A. Nature of TSPA Peer Review and this Interim Report

This is the first interim report of the Peer Review Panel. The Panel is scheduled to produce two additional interim reports during the completion of the TSPA-VA (see Appendix B). After the TSPA-VA is complete, the Panel will formally review the final TSPA-VA report and prepare a final peer review report. The Panel intends to follow a similar framework for each report.

This report reflects the Panel's initial understanding of the TSPA-VA, subject to several important limitations:

- The TSPA-VA has not yet been written and, thus, the Panel is reviewing a work in progress. The Panel has available to it previous TSPA reports and various technical reports prepared in support of TSPA-VA. The Panel also has been given briefings on the work being done for the TSPA-VA and has sent Panel members to "abstraction" meetings conducted by the TSPA-VA team to assign priorities to key technical issues and provide simplified versions of analyses suitable for an integrated analysis of the site as a whole. Members of the Panel have also attended other project workshops.

- The design of the repository is evolving. A number of aspects of the design have changed since the Panel's first meeting or are under review. Examples include the use of Alloy 625 instead of Alloy 825 as the inner corrosion resistant material for the waste canisters, the potential use of a gravel or sand backfill in the drifts, and consideration of alternative configurations for spent fuel containing highly enriched uranium.

- The Panel is still in an orientation phase. This is a large and technically diverse project with many years of history. In the few months since the Panel's first meeting, members have not had time to review all relevant documents or to learn of all the work that is being done to support the TSPA-VA. As a result, certain issues, such as seismic contributions to risk, gaseous pathways of exposure, the performance of spent fuel cladding, and the effects of volcanism and human intrusion, will be addressed in later reports.

For these reasons, this initial interim report addresses the overall scope and structure of the TSPA-VA and key physical processes that are anticipated to occur; it does not consider computational issues. In this initial interim report, the Panel is focusing on the analytical framework and conceptual models that make up the TSPA-VA. Before the Panel makes a detailed effort to determine whether the analyses are being performed correctly, it will be reviewing whether the project is addressing the right problems and is identifying the key tasks required for successfully completing the TSPA-VA.

These aspects will be evaluated in the context of their overall significance to a determination of the long-term performance of the proposed repository at Yucca Mountain, and against the Panel's judgments regarding the degree to which the TSPA-VA can provide a technically appropriate basis for assessing regulatory compliance. When forming the Peer Review Panel, the CRWMS M&O asked the Panel to consider the degree to which the analyses and supporting data underlying TSPA-VA are sufficiently robust to meet the "reasonable assurance" requirements of a regulatory review, in addition to reviewing the technical analyses of TSPA-VA on the basis of their scientific merit. While this is a question that can only be authoritatively answered by the U. S. Nuclear Regulatory Commission (USNRC), the Panel will comment on the general suitability of TSPA work as a basis for a regulatory review to help identify where additional data and analyses are needed.

This report includes three sections:

I. Introduction--outlines the nature of the peer review and of the TSPA-VA, describes how a repository is expected to work, and discusses assumptions about applicable standards;
II. Processes and events--describes the major processes and events that could lead to releases; and
III. Findings--summarizes major initial findings discussed in Sections I and II.

The organization of the technical content of Section II, Process and Events, is illustrated in Figure 1.

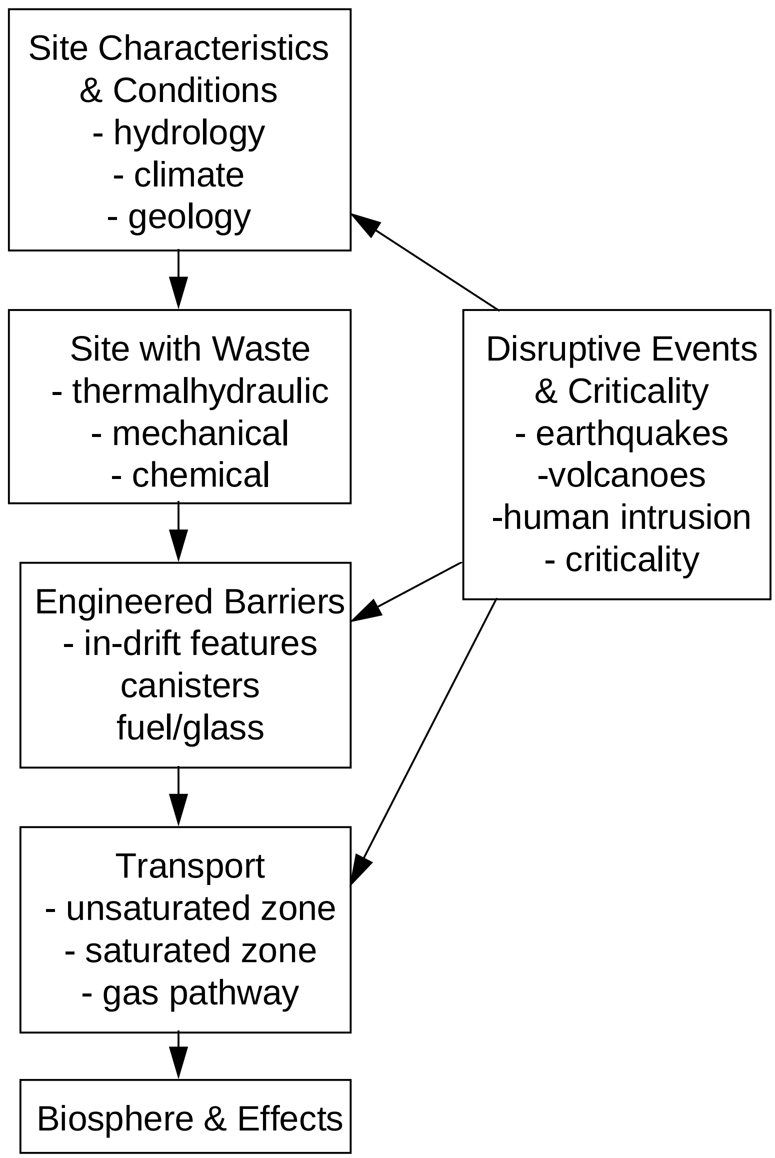

Figure 1 -- Organization of TSPA-VA Peer Review

B. Models and Uncertainties in TSPA-VA

Overview

Models with differing levels of detail are used for different purposes. Detailed process models, for example, incorporate as much of the complexity of a system as possible. Their purpose is to provide an understanding of associated physical processes at a fundamental level. Because of their complexity, such models are typically used to perform only a limited number of deterministic calculations.

A deterministic calculation is one in which all parameters are represented by single values, and in which the performance measures are calculated as point estimates. In contrast, in a probabilistic analysis, analytical parameters are represented by probability distributions and performance measures are estimated and expressed as probability distributions.

These probability distributions reflect both the uncertainties and variability in the associated factor or parameter. These two factors, uncertainty and variability, are treated in an analytically similar manner in a probabilistic analysis. However, they reflect different concepts. Uncertainty refers to a lack of knowledge about a factor, parameter, or model; variability refers to true heterogeneity or diversity in an exposure parameter (EPA, 1997). For example, the rate at which a specified waste package material corrodes in a specified environment is uncertain; the degree by which the permeability of the rock in Yucca Mountain differs from one location to another reflects its variability.

When a probabilistic model is run on a computer, the computer performs a large number of deterministic calculations, using parameter values selected on the basis of their specified probability distributions. It takes many such realizations (or cycles through a system model) to compute the statistical properties of the result sought. For this reason, simplified system or TSPA models, containing fewer details than deterministic models, are used. These simplified models can be run much faster (and, therefore, many more times) than the more complex deterministic models. Such system models are still expected to reproduce the essential physical processes of the repository system.

Neither process models nor system models can represent with complete accuracy the physical processes operating in a complex geologic system. Model calculations will include some degree of uncertainty. These arise from an inevitable lack of a completely detailed understanding of all the events and processes that are modeled, limitations of the models themselves, and the lack of or inaccuracies in the measurements or parameters, such as rate of water flow through fractures, that are used in the calculations.

A primary goal of a TSPA is to make the system or abstraction models yield results that are as realistic as possible. Where it is not possible to capture the complexity of the detailed models through abstraction, the goal is for the abstraction models to capture the essential characteristics of the system in order to bound the prediction of performance. This makes it imperative that those responsible for preparing the abstraction models have a clear and thorough understanding of the detailed process models. Model developers should ideally know about all the alternative models that have been developed, the strengths of each, and how they compare one with another. Use of the output of the alternative models provides insight into the distribution of the uncertainties associated with a specific feature, event, or process.

Analysts use more detailed process models to help identify phenomena and uncertainties that most affect estimates of various aspects of system performance. In other words, analysts use detailed process models to help them understand the phenomena and uncertainties to which those estimates are most sensitive**.** This information helps guide the next series of studies with the detailed process models. The events and processes that are found to be significant are then incorporated into the system models.

The system models are referred to in the TSPA-VA as "abstraction models" since the development of the system models proceeds by extracting the essential features from detailed calculations performed with the process models, that is, by abstracting those calculations. Constructing system or abstraction models is not necessarily a simple task. It requires a series of iterations and feedback loops between the detailed process models and the system models.

System and Abstraction Models

The TSPA-VA Plan (CRWMS M&O, 1996) enumerates a range of reasons for developing abstraction models. One of the most important of these is the inherent complexity of the performance assessment. If the analysis were to be conducted with detailed process-level models, the number of simulations that would be necessary would overwhelm existing computing resources.

In addition to the consideration of computational requirements, a probabilistic model may provide better insights into system performance than a more complex deterministic model when the variability of various factors is the key determinant of performance. For example, a deterministic model may be developed to estimate the time it takes for a waste canister to be penetrated by pitting corrosion. Under identical conditions of temperature, humidity, and water contact, such a model would predict the expected time to penetration for all canisters in the specified environment. However, a probabilistic model is better able than a deterministic model to represent small differences in both the environmental conditions and in the variability in canister performance due to manufacturing details. This permits a probabilistic model to project the distribution of the times to penetration for a large number of canisters, and avoids the unrealistic result that they all are penetrated at precisely the same time.

Often, the added capabilities of a complex deterministic model have little effect on predictions of overall system performance. It may be the case that, for a model that bases its calculations on hundreds of input parameters, only two or three parameters have a significant effect on the modeled results. In such a case, abstraction to a two or three parameter model simplifies a calculation without losing its essential features.

In addition, it may be the case that the factors that determine how some aspect of a repository would perform are sufficiently complex and poorly understood that, even without consideration of uncertainties and variabilities, development of a realistic deterministic model is beyond the current state of the science. In such cases, it may be possible to capture the available information regarding processes and performance directly in an abstraction model.

For detailed process models, a calculation typically includes only one component of the repository system. For studies using these detailed models, normally only a few representative values are used for any given parameter because of the time and cost required to obtain solutions. The results of these studies can be compared with laboratory or field data; they can be used to guide repository design; and they can be used for abstraction into system models.

For studies involving the system or abstraction models, the uncertainty in each key parameter is quantified by defining a probability density function (pdf). A probability density function is a function or curve that describes all the possible values of a parameter, weighted to reflect the relative likelihood of each value. A probabilistic model can use pdfs for parameters to estimate the uncertainties in the performance measures that result from the parameter and model uncertainties included in the analysis. The predictions generated by the system or abstraction models can also be used as a measure of the sensitivity of the performance measures to various input parameters.

Before any abstraction model is used, the results it generates must be compared to those produced by the more detailed model and be shown to be reasonable and conservative, that is, it overestimates rather than underestimates the risk. While the ideal for TSPA-VA is a realistic estimation of performance, often this is not possible, and a conservative or bounding analysis is used. The abstracted model should be used only if the process-level model confirms that its use is justified.

Separate groups of analysts are working on particular parts of the TSPA-VA, for example, on analyzing waste form alteration and mobilization or the behavior of the near-field environment. A key to success of the TSPA-VA will be to integrate properly the abstraction models developed by the separate groups.

C. Policy and Regulatory Framework for the Repository Analysis

The Nuclear Waste Policy Act of 1982 (P.L. 97-425) and the Nuclear Waste Policy Act Amendments of 1987 establish a process and assign responsibility for developing a repository for the disposal of high-level radioactive waste (HLRW). Under these laws, DOE is charged with investigating a site at Yucca Mountain, Nevada, to determine its suitability for a repository. If, after studies of the proposed site have been completed, the responsible DOE staff members conclude that the site is suitable, their recommendation must be reviewed and upheld through a specified process. If it is, the staff will prepare and submit a license application to the USNRC for authorization to construct the repository. The USNRC is responsible for reviewing the application and determining whether the facility can be constructed. After construction, the USNRC must issue a license before the facility can be operated. Up to 100 years after emplacement operations begin, the repository will be closed after the USNRC has issued the necessary authorization.

By law, no more than 70,000 metric tons of HLRW can placed in the first repository until a second repository is operational. Under current policy, about 90% of this amount (63,000 metric tons) will be spent commercial fuel and the remainder (7,000 metric tons) will be defense HLRW, including liquid high-level waste that has been incorporated into glass or vitrified (DOE, 1997). Additional DOE owned wastes now expected to be included for disposal include Navy reactor fuel, DOE fuel from research reactors, and, possibly, waste forms resulting from the immobilization of excess weapons plutonium.

At the present time, the standards that the USNRC will apply to determine whether a repository can be safely operated have not been established. Section 801 of the Energy Policy Act of 1992 (P.L. 102-486) directs the U. S. Environmental Protection Agency (EPA) to promulgate standards to protect public health as a result of activities related to the construction, operation, and closure of the proposed repository at Yucca Mountain. The EPA is to base the standards on the results of a study by a committee of the National Academy of Sciences/National Research Council (National Research Council, 1995a). Following establishment of the EPA standards, the USNRC is to promulgate regulations, consistent with the protection requirements established by EPA. At the present time, EPA has not yet published a proposed standard.

Preparation of the TSPA-VA presents a significant technical challenge. Exacerbating this challenge is the lack of promulgation of the standards or regulations applicable to the proposed repository. Until such time that the EPA and the USNRC complete these actions, many aspects of the work on the TSPA-VA will continue to be conducted in a regulatory vacuum.

D. Communicating the Repository Concept and How It Is Intended to Work

In the Peer Review Plan (see Appendix B), the Panel is asked to consider not only the analytical approach of the TSPA-VA, but also its traceability and transparency. Although the Panel agrees these are important considerations, since the TSPA-VA has not been written, the Panel cannot yet comment on the TSPA team’s success in achieving the important goals of traceability and transparency.

Recently, a subgroup of the Performance Assessment Advisory Group of the European Nuclear Energy Agency reviewed specific technical approaches in ten recent system assessments including two addressing Yucca Mountain, one by DOE and one by the USNRC (NEA, in draft). The subgroup noted problems in the reports with clarity, readability, and completeness, that is, in traceability and transparency. The subgroup defined traceability as a complete and unambiguous record of decisions and assumptions and of models and data and their use in arriving at results. In their view, traceability in documentation is achieved if an independent performance assessment group can reproduce the reported analyses. Transparency, according to the subgroup, requires ensuring completeness and using a logical structure that facilitates in-depth review of the relevant issues. It is achieved when a reader or reviewer has a clear picture of what was done in the analysis, what the outcome was, and why. The degree of transparency of a document to a particular reader will vary by the technical background of the reader and the purpose for which the document is being read. Thus, transparency depends in part on accurate identification of the audience and its needs.

In the Panel's view, a key aspect of achieving traceability and transparency is to begin by communicating the overall description or conceptual model of the repository and of the processes and events to be analyzed. It is helpful for any reviewer to understand what is being analyzed before reviewing how it is analyzed Thus, documentation should first convey how the repository is intended to work and the processes that are to be modeled before discussing how the processes have been modeled. The Panel recommends that the overall conceptual model of repository performance and the time sequence and interactions of possible future events be described in an introductory chapter of the TSPA-VA.

The Panel also recognizes its own obligation to communicate its findings and reasoning completely and clearly to the reader. As noted in the Preface, the Panel is writing for two audiences, one of whom may require background information about what a repository is and how it is intended to work to protect people and the environment before reading about the TSPA-VA. The following sections provide a general description of a repository and the current design and discuss how the functioning of the whole system changes over time.

Description of a Repository and the Current Design

The basic idea behind the design of the repository is a simple one:

- The wastes are to be contained within an engineered barrier system (which includes the waste form, fuel cladding or vitrified waste packages, canisters, possibly backfill, and the drift liners) for as long as feasible to permit decay of a large portion of the radioactive material in the waste. Further, the engineered barrier system is to function so as to provide for the slow release of the remaining radionuclides over a long period of time.
- The geologic formation in which the repository is placed is expected to provide additional barriers to exposure of people and the environment through natural retardation of the movement of the radionuclides.
- Once the radionuclides reach the groundwater, a long flow path toward the environment is expected to allow additional time for the radioactive material to decay and to dilute the radionuclides.

Key elements of the engineered barriers envisioned by the current design are:

- a hot, dry zone to prevent liquid water from coming into contact with the waste packages, [The design being considered in the TSPA-VA envisions an initial period during which the heat generated by the waste creates a local zone around the repository that exceeds the boiling point of water. Corrosion of the waste canisters would occur slowly as compared to the rates when water as a liquid is present.];
- a concrete drift liner which, possibly in conjunction with backfill, would protect waste packages from falling rock, such as might result from an earthquake;

- a two layer waste package or canister, made up of an outer, corrosion allowance material and an inner corrosion resistant material; and
- the waste form, either spent nuclear fuel or glass logs.

Following the hot, dry period, water will reach the waste canisters and corrosion will eventually breach the canisters. Water can then enter and leach radionuclides from the spent fuel and vitrified waste. Moisture and radionuclides will be transported from the altered waste through the remainder of the canister and barrier system to the geologic system. This water, with accompanying radionuclides, would then be transported downward through the unsaturated zone, that is, the area above the water table or saturated zone. Some degree of geochemical retardation is anticipated to occur during movement through the unsaturated zone. Radionuclides would eventually reach the groundwater in the saturated zone and would be transported in the groundwater to a point at which the water might be extracted for human use. Physical features such as the presence of fractures or of perched water tables, that is, isolated pockets of saturated rock, will vary throughout the site. These variations in the physical features or heterogeneities will lead to a distribution in the time at which canisters fail at different locations and a variation in the rates of movement of any released radionuclides through the unsaturated zone.

If the above processes occur as anticipated in the design, the movement of radionuclides in the repository to groundwater would be delayed long enough for a significant amount of radioactive decay to occur, and the remaining radionuclides would enter groundwater over a long period of time, so that significant dilution can occur.

This is the picture of repository performance that the TSPA-VA must address in a quantitative manner. The key analytical issues are to identify the important process, determine when these processes occur, and their rates.

Sequence of Events

While the results of many assessments of repository performance, and associated sensitivity analyses, were presented in the 1995 TSPA (TSPA-95) (CRWMS M&O, 1995), these results would be more easily understood if they were accompanied by a description of how the performance of the repository evolves over time. The key element of such a description is the likely state of the repository and waste at future times. Given the large number of processes and barriers, graphical presentation of results would be helpful. The description should include as a function of time the:

- conditions in the unsaturated zone above the repository;
- composition of the waste, taking decay into account;
- duration of the hot, dry period, and associated humidity conditions;
- changes in the liner and in-drift chemistry;
- chemical conditions and vapor/liquid water reaching waste packages during and after the hot, dry period;
- corrosion penetration rates and progression of damage through the outer corrosion allowance material;

- corrosion penetration rates and progression of damage through the inner corrosion resistant material;
- physical and chemical processes in the canisters that lead to degradation of spent fuel and vitrified waste;
- rates of mobilization of radionuclides from the waste packages;
- processes and rates of retardation of radionuclides in the near-field;
- travel times through the unsaturated zone, considering alternative flow paths, retardation, and breakthrough;
- dilution and movement of radionuclides in the saturated zone;
- behavior and movement of released radionuclides within the biosphere and avenues of exposure and uptake by nearby population groups;
- assumed locations and lifestyles of nearby populations; and
- assumed values for various factors used to translate radionuclide concentrations into dose and risk estimates.

The point of such a description is to communicate the time delay and degree of isolation that each element of the repository can afford, and to describe the sequence and timing of events that could lead to radionuclide releases and human exposure. Due to the complexities noted in the following discussion of sensitivities, the Panel recommends that this time-based overview of site performance be provided without reference to a specific regulatory time period. The questions of "How would the repository perform?" and "Does it comply?" are separate.

Impact of time sequence on sensitivities

The useful sensitivity analyses in the TSPA-95 (CRWMS, 1995) and subsequent presentations to the Panel indicate that two factors above all others affect the TSPA results: (1) whether the applicable standard is based on performance over 10,000 years or until the time of peak dose or risk (up to 1,000,000 years), and (2) the rate at which water enters the repository--the infiltration rate of water. What is particularly notable about these two factors is that, depending on the values selected, there is a change in the key factors that determine whether the predicted performance of the proposed repository is acceptable. For example, if the standard applies for 10,000 years, the duration of the hot, dry period and subsequent rates of waste package corrosion and unsaturated zone transport are the primary factors in the analysis. However, if the standard applies to peak doses or risks at any future time, the details of the duration of the hot dry period and many associated analytical issues concerning corrosion rates become less important. For this peak dose case, the infiltration rate and rate of dilution in the saturated zone are major considerations.

Similarly, the amount of water that annually moves through the repository horizon affects the shape and duration of the hot, dry zone and, subsequently, the time it takes for canisters to fail and the rate at which radionuclides are transported. As is noted in Section II, for low infiltration rates, the analyses have indicated a repository-wide hot, dry zone. For higher infiltration rates, there may be regions between adjacent drifts that do not exceed the boiling point. As a result, the nature and rate of unsaturated zone flow depend strongly on the infiltration rate.

Impact of time sequence on performance measures

The TSPA-95 evaluates performance in terms of individual peak doses received within 10,000 and 1,000,000 years. The use of these two time points is reasonable, since the actual standard that will apply to Yucca Mountain is not known. The 10,000 year period is the duration specified in 40 CFR 191, the existing EPA standard applicable to WIPP, and the 1,000,000 year period is consistent with recommendations from the National Research Council Committee on Technical Bases for Yucca Mountain Standards (National Research Council, 1995a).

There may be situations in which these two time points may not provide sufficient insights into the performance characteristics of interest. For example, it appears likely that engineered barriers will have little effect on performance measured against a 1,000,000 year time period, so that analyses against this potential standard are uninformative regarding the incremental contributions to waste isolation afforded by the engineering measures under consideration. Conversely, many if not most analytical realizations conducted for a 10,000 year period may indicate that no exposures will occur within that period. (The point at which compliance will be determined is itself an uncertain feature of the applicable regulations). In this latter case, all design alternatives may appear capable of producing regulatory compliance. Little information about the comparative value of design approaches would be obtained. Alternative measures of performance, for example, the distribution calculated for the time for contaminated groundwater to reach the point of compliance, would be more informative.

The prediction of the performance of a repository at Yucca Mountain is a technically challenging task. As is noted in Section II below, the analysis involves modeling complex hydrologic, thermal, chemical, and mechanical interactions over times spanning tens or hundreds of thousands of years. Many of the technical questions that are necessarily addressed in this analysis, for example, the distribution of the time for corrosion to breach the waste canisters, are at the forefront of the state of the science. It is with these complexities in mind that the Panel has evaluated the TSPA-VA work that has been done and that is in progress.

II. PROCESSES AND EVENTS

This section describes the Panel's overview of (1) the site characteristics that are important to the performance of a waste repository, including an analysis of the site under the thermal load that the radioactive waste in the proposed repository would create, (2) the nature and effect of various engineered barrier systems, (3) the waste form and its anticipated degradation over time, (4) disruptive events, including criticality, (5) transport of radionuclides in the unsaturated and saturated zones, and (6) the biosphere, radionuclide transport, the accompanying doses to exposed population groups, and the resulting risks. The organization of these topics is illustrated in Figure 1, above.

The Panel's purpose for including the following technical discussion in this first interim report is to provide the TSPA-VA team and others with the Panel's initial understanding of the important issues in assessing how the site and repository design would perform. It is the Panel's intent that this will lead members of the TSPA-VA team and others to call to the Panel's attention studies and issues that may have been overlooked or misinterpreted. As noted in the introduction to this report, the Panel is still in an orientation stage with regard to the large amount of work that has been done and the volumes of literature to review relevant to the content of the TSPA-VA.

A. Initial Conditions

Overview

The nature and characteristics of the geologic environment at Yucca Mountain involve a rock system with a complex set of initial conditions. The characteristics of this rock system in its initial state must be understood, so that the effects of a disturbance to these initial conditions caused by the construction and thermal effects of a repository can be assessed.

For the past 15 years, DOE has carried out a comprehensive series of studies, both in the laboratory and in the field, that provide a wide range of scientific and technical information on the geologic nature of the site. The following discussion presents a topical review of the information that the Panel believes is vital to an analysis of the supporting data underlying the TSPA-VA.

The following sections include:

- a brief description of the rock formation at Yucca Mountain to provide an idea of the geologic setting of the proposed repository. The repository would be in a sequence of unsaturated layers of volcanic tuffs.

- a discussion of a conceptual model that has been developed for analyzing this unsaturated system. Incorporated into this model are a number of important processes with characteristics and behavior based on available information.
- a discussion of a second conceptual model for the saturated system beneath the repository which is designed to permit analyses of the potential migration paths to the biosphere.

One important consideration for the TSPA-VA team is determining which aspects of the proposed repository setting are fundamentally important to the development of an acceptable performance assessment.

Geologic Setting

Yucca Mountain is a mountainous ridge in the southern Great Basin, the largest subprovince in the Basin and Range province of the United States. This province is characterized by more or less regularly spaced subparallel mountain ranges and intervening alluvial basins formed by extensional faulting. The strategy for waste isolation at Yucca Mountain (YM) is to rely on a number of barriers, both natural and engineered, that either are attributes of this site or are engineered in a manner to complement its attributes.

The subsurface hydrologic processes at YM occur in a system of heterogeneous layers of anisotropic, fractured volcanic rocks in an arid environment. The mean annual precipitation is approximately 170 mm per year, but exhibits considerable variation from year to year. The present-day climate at Yucca Mountain probably has existed for less than 10,000 years. The likely pluvial conditions at Yucca Mountain were characterized by average annual temperatures approximately 6° to 7°C cooler than present and winter rainfalls approximately 60% to 70% higher than at present (Spaulding, 1985).

The volcanic rocks at YM consist of alternating layers of welded and nonwelded ash flow and air fall tuffs. The primary geologic formations at YM, from the surface downward, are the Tiva Canyon, Yucca Mountain, Pah Canyon, and Topopah Spring Tuffs that are known as the Paintbrush Group. Beneath this group are the Calico Hills Formation and the Crater Flat Group, which includes the Prow Pass, Bullfrog, and Tram Tuffs (Buesch et al., 1995). Each of these formations is underlain by an associated, nonwelded bedded tuff layer. These geologic formations have been divided into five hydrogeologic units (Montazer and Wilson, 1984) based roughly on the degree of welding: (1) the Tiva Canyon welded (TCw); (2) the Paintbrush nonwelded (PTn) (consisting primarily of the Yucca Mountain and Pah Canyon members and their bedded tuffs); (3) the Topopah Spring welded (TSw); (4) the Calico Hills nonwelded (CHn); and (5) the Crater Flat undifferentiated (CFu) units.

The welded units typically have low matrix porosities (8%-12%) and high fracture densities, whereas the nonwelded and bedded tuffs have relatively higher porosities (20%-45%) and lower fracture densities (Montazer and Wilson, 1984). At smaller scales, the fracture density is correlated with increases in the degree of welding. Some parts of these units can be altered to zeolites or clays depending on their cooling history and the presence of groundwater and heat.

Alteration does not affect porosities significantly, but it does decrease permeabilities of the formations where it occurs.

Conceptual Model for the Unsaturated Zone

A three-dimensional conceptual model of the UZ at Yucca Mountain is being developed by Lawrence Berkeley National Laboratory (LBNL) in collaboration with the U. S. Geological Survey (USGS). Work on this model was initiated several years ago, and there have been a number of subsequent modifications. The 1996 version is described in detail by Bodvarsson and Bandurraga (1996)[1] and is the primary source for the description of the conceptual model given below.

Figure 2 is a schematic cross section of YM that shows the layering of the hydrogeologic units described above and presents a pictorial summary of several important aspects of the conceptual model. The Solitario Canyon fault on the west edge and the Ghost Dance fault in the center of the section enclose a large emplacement block of Topopah Spring tuff (TSw) in which the proposed repository is to be constructed. A second smaller block of TSw on the east side of the Ghost Dance fault will be used to expand the repository as necessary. The water table is 570 m below the surface and, as can be seen on Figure 2, the location of the repository is about halfway down to the water table. The following discussion presents summaries from Bodvarsson and Bandurraga (1996) of several important components of the conceptual model shown on Figure 2.

Infiltration

Infiltration at YM is spatially heterogeneous due to variations in soil cover and topography. It is also variable with time due to storm events (Hevesi et al., 1994). The current conceptual model for infiltration is based on numerous measurements of water content profiles in shallow boreholes. A significant thickness of alluvium can store infiltration and attenuate an infiltration pulse. Thus, infiltration is high on sideslopes and ridgetops where outcrops are exposed and flow into the fractured volcanics can take place (Flint and Flint, 1994). To avoid uncertainties associated with temporal variations, infiltration can be characterized with maps that provide steady-state infiltration rates (Hevesi and Flint, 1996; Flint et al., 1996). Spatially variable infiltration maps have been developed by project scientists, and temporally variable maps showing historic variations in precipitation due to pluvial periods have also been developed.

The magnitude of the infiltration rate used in modeling investigations has covered a wide range. Much of the early characterization work assumed infiltration rates that were around 0.1 mm per year, but this has recently been increased to 5 mm per year for the current “dry” climate. In the case of a wetter climate from anticipated pluvial periods in the future, it has been estimated that infiltration rates may range from 10 mm per year to as much as 30 mm per year. This is a critical

[1] An LBNL report on an updated version of this UZ model is scheduled to be issued in June 1997.

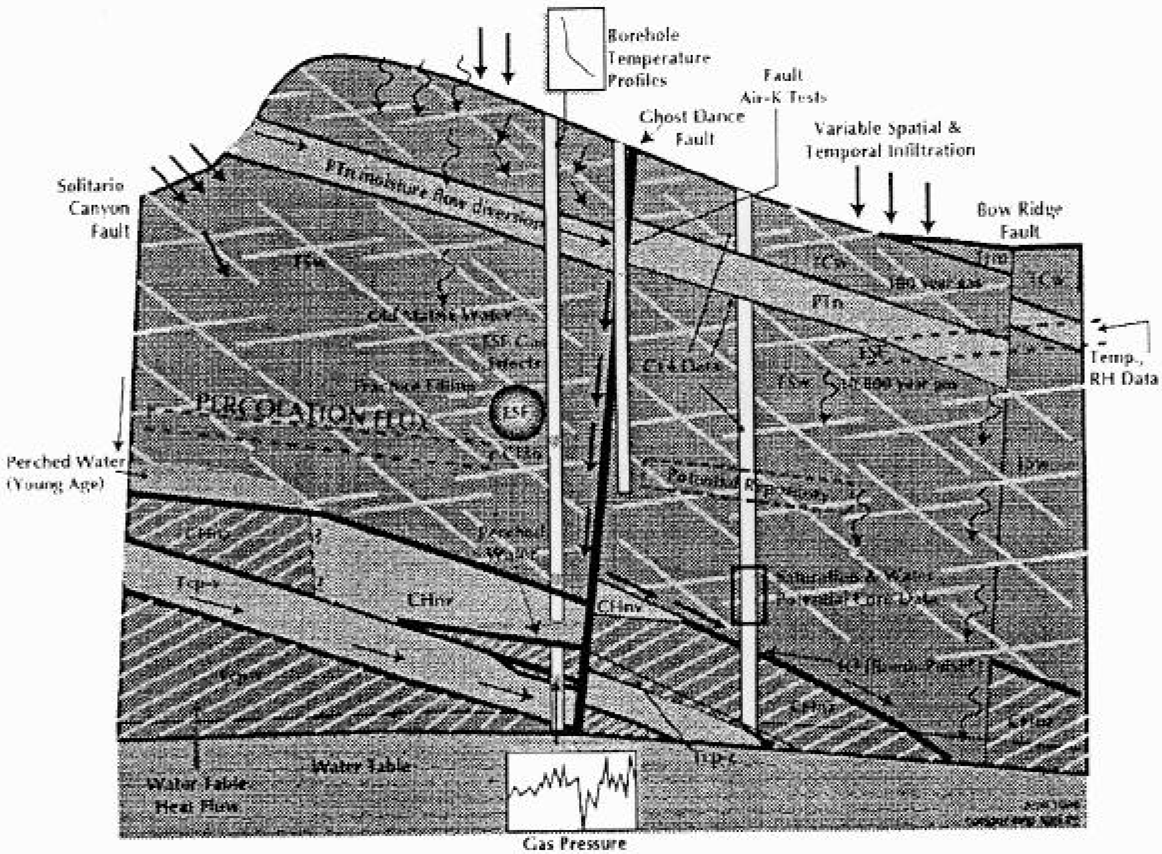

Figure 2 -- West-East schematic cross section through Yucca Mountain showing various data and processes of the conceptual model (after Bodvarsson and Bandurraga, 1996)

factor, because the percolation flux at the repository level depends on the magnitude of the infiltration rate.

Geothermal gradient

The geothermal gradient at YM is relatively well-established through numerous temperature and heat flow measurements that have been made by Sass et al. (1988). In analyzing recent borehole temperature profiles, Rousseau et al. (1996) incorporated an analysis of the data which linked variations in the geothermal gradient to infiltration rates, and thus allowed one to obtain an appropriate percolation map for the proposed repository region. The concept is that as the water of infiltration migrates downward, any increase in the percolation flux that develops will cause a slight reduction in formation temperatures. This changes the variation of temperature with depth, which can then be compared to measured data for the geothermal gradient.

Assuming reasonable heat flux values, one obtains a percolation flux through the repository horizon in the TSw of some 5 to 10 mm per year (Bodvarsson, Wu, and Finsterle, 1996). Analysis of temperatures in the CHn and deeper units shows, in general, less percolation flux; this suggests lateral flow above this part of the section, perhaps due to the low permeability zeolites, as will be discussed below. However, the sensitivity of this method to distinguish between percolation fluxes in the range of 5 to 10 mm per year needs to be confirmed.

Percolation flux

The percolation flux at the level of the proposed repository in the middle of the non-lithophysal portion of the TSw (Fig. 2) is one of the most critical parameters both in interpreting the current conditions at the site and in assessing its suitability as a potential repository. Presumably, this flux has led to the present distribution of water saturations in the matrix of the TSw, which range from 50% to 70% in the top half of this layer and up to 90% to 95% in the bottom half.

Data from several different field observations suggest that the flux reaching the repository level, by migration through the matrix, is relatively small, which implies that fracture flow is the dominant factor. These data include: (1) relatively low matrix saturations in the upper portion of the TSw, which retard or preclude downward movement (Flint, 1996); (2) lack of observance of weeping from exposed surfaces during construction of the Exploratory Studies Facility (ESF) (Rousseau et al., 1996); and (3) the fact that if fracture coatings are continuously deposited on the fracture-matrix interface, the percolation flux moving through the system is very low (Peterman and Paces, 1996). In addition, data suggest that the liquid flux may reach the repository level mainly through episodic fracture flow (Burger and Scofield, 1994). Additionally, an analog site at Rainier Mesa with an average annual precipitation approximately double that of YM is reported to have 23.7 mm per year infiltrating deep into the subsurface through fault or fracture flow (Wang et al., 1993).

Lateral flow

Lateral flow is depicted on Figure 2 because it has an important effect on the spatially variable infiltration rate and contributes to the diminution of the actual percolation flux that reaches the level of the repository. A capillary barrier is believed to exist between the welded TCw and the underlying PTn which prevents infiltration into the PTn (Moyer and Geslin, 1995). A capillary barrier to the imbibition of a wetting phase, such as water, would be a jump from a low to a high permeability medium. A flow barrier would be one of reduced flow conductivity. This distinction also needs to be made below, in the discussion of faults.

The capillary barrier leads to lateral diversion of moisture through the TCw fractures and migration down the dipping geologic layers until a fault or fracture intercepts the flow and provides for downward movement. Rather than having an abrupt and laterally continuous barrier between the TCw and PTn, a gradational contact has been found in recent studies. This presents a problem in setting up the appropriate representative numerical system for modeling this kind of heterogeneity.

Major faults

The effects of major faults are yet another feature that must be incorporated into models of YM in an effort to develop a reliable picture of this complex system. The proposed repository area is surrounded and crossed by numerous strike-slip and normal faults with varying amounts of offset (Scott and Bonk, 1984). The previously mentioned Ghost Dance fault, a North-South near vertical fault with segments of the potential repository on both sides (Fig. 2), has offsets of tens of meters (Spengler et al., 1993) while the Solitario Canyon fault located to the west has offsets of up to hundreds of meters. Modeling studies have been done assuming the faults to be either capillary barriers to the imbibing water or hydraulic permeability barriers (Wittwer et al., 1995); there is evidence for both situations.

Considerable insight into flow characteristics of the rock mass at YM has been gathered from measurements of gas pressures at varying depths in boreholes (Fig. 2). As atmospheric pressures fluctuate, the pneumatic effects propagate downward causing changes in these subsurface gas pressures that can be analyzed. Pneumatic data obtained from sensors deep in the TSw show that the Drill Hole Wash fault, which lies to the north of the cross-section in Figure 2, appears to block the effect of pressure disturbances moving through the ESF (Ahlers et al., 1996). It is conceivable that this gas flow might be blocked by low permeability gouge in the strike-slip fault. Alternatively, this fault may consist of a high porosity zone which attenuates the signal from the ESF in much the same way as the PTn attenuates the atmospheric signal transmitted through the TCw. Studies in Alcove 2, which is an opening off the ESF, indicate that the air permeability of the Bow Ridge fault (Fig. 2) is about 2E-11 m^2 (Bodvarsson, Bandurraga, and Wu, 1996), or about the same as that of permeable bedded tuff or highly fractured welded tuff.

Additional evidence supports the interpretation that the faults may serve as hydraulic permeability barriers. For example, a water body observed at borehole SD-7 is thought to be perched over a zeolitic layer and prevented from lateral movement by the presence of the Ghost Dance fault (Fig. 2). Rousseau et al. (1996) have presented a similar interpretation for a perched water body intercepted by a borehole not too far from the Solitario Canyon fault.

Another source of evidence for faults acting as permeability barriers is the potentiometric surface measurements that have been made at YM. At a location that is somewhat to the north of the section on Figure 2, the elevation of the potentiometric surface on the western side of the Solitario Canyon fault is approximately 40 m higher than the elevation of this surface on the eastern side (Tucci and Burkhardt, 1995). This gradient could only be maintained if the fault is a hydraulic barrier to flow.

Perched water

The presence of perched water has implications for the travel times and flow paths of water through the UZ. Several pumping tests conducted in borehole UZ-14, which lies to the north of the section on Figure 2, showed a quick pressure response indicating that there is an extensive perched-water body around the area of this borehole (Burger and Scofield, 1994). Wet layers have also been found in a number of other boreholes, and these occurrences have been detected in zones in, or overlying, geologic units which have been extensively altered to zeolites. This condition is often encountered in the CHn material underlying the densely welded and fractured TSw (Fig. 2).

Perched water occurs where abrupt permeability differences exist between adjacent formations. Such an occurrence often results in the presence of zeolites, as is evident in the zeolitic permeability data measured on borehole samples. Permeabilities range from 1E-18 m^2 down to the lower limit (1E-21 m^2) that the available instrumentation could measure (Flint, 1995). Where lateral movement is prevented, the low permeability of the altered material is one reason for the formation of perched water bodies above such layers.

Zeolites

The effect of zeolites on flow paths is another important component in the conceptual model of the UZ below the repository. Where barriers to lateral movement at faults do not exist, the extensive development of zeolitized layers in the CHn and possibly in certain layers in the CFu can lead to an easterly migration of moisture downslope to the water table (Fig. 2). This is critical to the problem of radionuclide transport because the path of migration beneath the repository, which may be vertically downward initially, can be subjected to extensive diversions as a result of lateral flow.

Fast fluid flow paths

The presence of fast fluid flow paths is another critical aspect of the conceptual model at YM because of the implications for rapid transport of radionuclides through the system. One method of identifying their existence is through age analysis of isotopes of various elements. A number of isotopic measurements are being performed on perched water samples, matrix pore water samples, and matrix samples obtained from the UZ. In addition, fracture coatings and fillings have been tested for isotopic concentrations. These measurements can provide an estimate of the age of the samples, if the historic atmospheric isotopic concentrations and other factors are sufficiently known. Recent work on fracture coatings from the ESF have not shown any age estimates younger than about 54,000 years, with a majority of the samples in the range of 90,000 - 200,000 years (Paces, 1996). However, the mechanisms by which the fracture coatings are deposited are not completely understood, and these results may be biased toward longer ages.

Some of the recent matrix samples taken from the ESF and in boreholes show the presence of elevated concentrations of ^{36}Cl in the PTn and upper TSw (Fabryka-Martin and Liu, 1995; Fabryka-Martin et al., 1996). One hypothesis for the elevated ^{36}Cl signals is that they are due to bomb-pulse infiltration, indicating relatively fast transport from the surface compared to matrix flow. One possible mechanism for this phenomenon is that it is associated with transient episodic flow through localized structural features which can penetrate into the TSw and then migrate into the matrix through fractures.

Preliminary data from isotope testing of perched water samples collected in boreholes indicate that the water has about 28% modern ^{14}C, with an apparent residence time since infiltration of 10,800 years (Bodvarsson, Bandurraga, Wu, 1996).

Conceptual Model for the Saturated Zone

The development of a conceptual model for the saturated zone (SZ model) at Yucca Mountain is not nearly as advanced as the UZ model. A large hydraulic gradient to the north of the potential repository has been identified as a key problem. Because this gradient is located upgradient from the repository, its presence may or may not be relevant to the performance of the repository. It is difficult, however, to argue that an adequate understanding of processes in the SZ exists without considering plausible alternative conceptual models of this feature. Large changes in the magnitude of the hydraulic gradient of the SZ seem to be associated with some structural features, but the relationship between the faults and SZ flow is highly interpretive. The work of Sass et al. (1988) on variations in temperature at the water table and geothermal flux indicates that there may be significant vertical flow in the SZ, and inferences along these lines from hydrochemical data may be useful in bounding the behavior of the system. This has important implications for the amount of dilution by vertical dispersion.

The distribution of recharge for regional-scale flow modeling, which predicts little or no recharge in the repository area, has been derived from a modified Maxey-Eakin method (D'Agnese et al., 1996). However, recent estimates of infiltration at YM (Hudson and Flint, 1996) suggest that there may be significant recharge to the SZ in some areas at the site. There is significant uncertainty regarding the ultimate discharge areas of flow from the region of YM. Alternative modeling studies indicate that the ultimate discharge points may be Franklin Lake Playa or Death Valley (or a combination of the two). It is possible that radionuclides transported to these discharge areas may interact with the biosphere by plant and animal uptake at springs or by wind erosion of playa surfaces containing precipitated material.

The question of flow channelization in the SZ is significant at a number of scales. It can occur in channels with lengths of only a few centimeters, or a few meters, or over hundreds o meters, and bears directly on the problem of dilution. The most important impact on the downgradient radionuclide concentrations in the SZ could come from flow channelization along large-scale discrete structural features. Such features may correspond to continuous, mappable structures at the surface, such as major faults. A likely candidate for such a flow feature would be zones of relatively continuous brecciation and tensile faulting, as has been observed in the area of the south ramp of the ESF (Day et al., in review).

The effects of wetter climatic conditions on the water table elevation beneath the repository have largely been inferred from isotopic data and from paleospring deposits. These inferences, as well as estimates of changes in SZ flow rates and patterns and changes in discharge areas, remain somewhat uncertain.

B. Site Conditions with Waste Present

Overview

The emplacement of some 70,000 MTU of spent fuel and vitrified high-level waste in the repository will have a major impact on the surrounding rock system. A large mass of water in the rock matrix immediately surrounds a facility of this magnitude. Predicting the overall behavior of the total system as it becomes thermally activated is a complex problem. Such predictions require the analysis of the hydraulic, chemical, and mechanical response of the rock layers as they are subjected to a thermal field that will be in a transient mode for thousands of years.

One of the key unresolved problems is the effect of the response of the rock system to the thermal field on the infiltration rate. Modeling studies indicate that for an infiltration rate on the order of 5.0 mm per year or more, peak temperatures at the repository level reach a boiling condition of 97° C near the repository that persists for about 1,000 years. Convective heat transfer driven by countercurrent vapor-liquid flow dominates in this constant-temperature boiling zone.

In contrast, when the infiltration rate is relatively low, such as 0.1 mm per year, the boiling zone is extensive and dry-out conditions develop and last for about 2,000 years. Presumably, during the dry-out period, there will be little, if any, water in contact with the engineered barrier. Thus, it is clear that the magnitude of the infiltration rate is a critical factor in determining when the engineered barrier will first be subjected to degradation by water that has been heated by the radioactive waste.

One of the most important considerations in the expected behavior of the repository is the fracture-matrix interactions that determine flow rates in the matrix and the fractures, and the resulting convective heat transfer. The ability to successfully model and assess the thermohydrologic conditions depends to a large degree on the understanding of the flow and displacement in this complex fracture-matrix system. A key factor is the role of capillarity in determining when fluids flow in the matrix or in the fracture.

The thermochemical response could be an important factor depending on the degree to which mineral deposition in fractures and matrix pores develops and there is a reduction in permeability. The thermomechanical response could be another important factor depending on the degree to which the fractured rock mass undergoes thermally induced displacements and changes in stress with a consequent reduction in permeability. These coupled effects, driven by the thermal load, are relatively close to the repository; and for the most part, they have been considered by the project to be of minor importance.

It is the Panel's view that the basis for the assumptions used in the determination of the infiltration rate that is finally adopted should be clearly stated in the TSPA-VA. The Panel is concerned that the mechanical response of fractures during the heating and cooling phases of the repository could significantly influence the fluid flow field in the vicinity of the waste containers. The Panel believes that the coupled (thermo-hydro-chemical-mechanical) behavior of the repository needs to be examined in order to correctly plan and interpret measurements from the proposed *in-situ* tests, such as the Drift-Scale Test.

Effects of Thermal Field on Hydraulic Behavior

Current plans for the proposed repository call for spent fuel and vitrified high-level waste to be enclosed in 11,000 containers of 5½ to 6 ft in diameter. The containers are to be distributed in 120 miles of tunnels and drifts that are to be constructed in the TSw with diameters of 15 to 20 ft. The emplacement drifts are to be mined 22.5 m apart, and the containers in each drift spaced to provide an areal loading with an average of 83 metric tons of initial heavy metal (HFU) per acre. The waste which will be emplaced over a period of 20 years and the emplacement area will cover 840 acres.

Various workers have studied these thermal effects using models that focus on the thermohydrologic behavior of a large global system enclosing the repository. The site-scale model that is being developed by LBNL and USGS (Bodvarsson and Bandurraga, 1996) has been designed to be large enough so that it can be used to investigate the global system surrounding the proposed repository. Haukwa et al. (1996) used a two dimensional (2-D) North-South cross-section of this model in thermal loading studies of the UZ zone. They used infiltration rates of 0.1 and 4.4 mm per year to investigate the effect of this factor on long-term thermal response.

Vertical temperature profiles through the center of the repository block over a period of 25,000 years are shown in Figure 3 for the case of 4.4 mm per year. The elevation of the repository in this figure is approximately 1090 m. Immediately apparent is the deviation in the thermal response of the system from the geothermal gradient over periods of thousands of years. Note that the peak temperature on the level of the repository reaches a boiling condition of 97°C at 100 years and remains there for about 1000 years. This simulated boiling condition extends over a vertical interval of about 100 m for 500-1000 years. Convective heat transfer driven by countercurrent

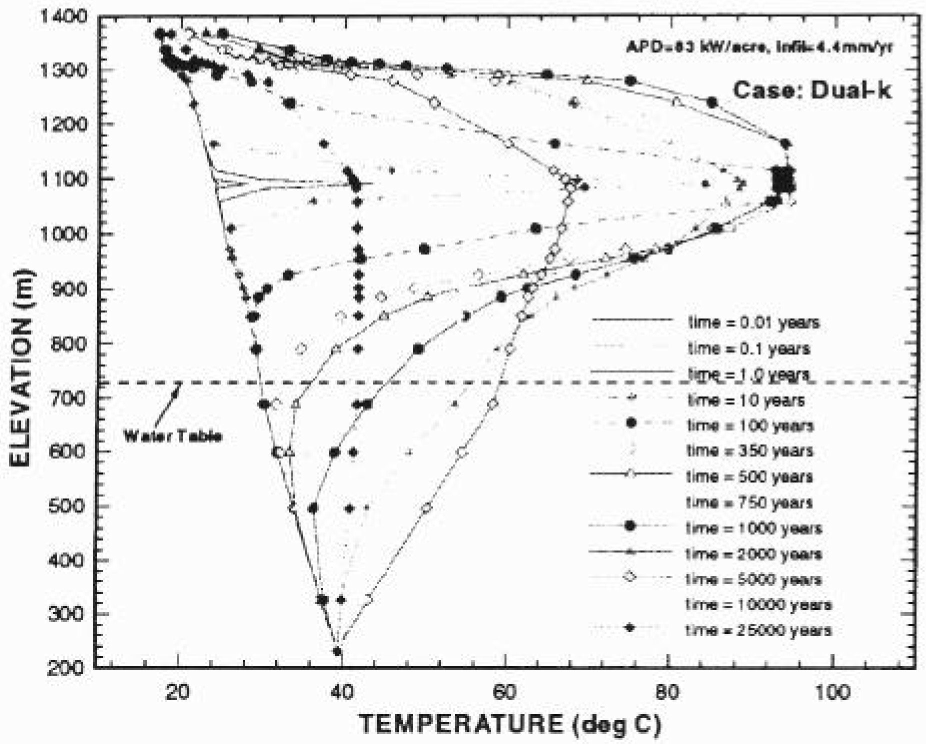

Figure 3 -- Vertical temperature profiles through the center of the repository block during 25,000 years of thermal loading at 83 kW/acre with Dual-k model and 4.4 mm per year infiltration (after Haukwa et al., 1996).

vapor-liquid flow dominates in this constant temperature boiling zone. Note also the temperature rise at the ground surface as well as in the saturated zone below the water table.

However, it is important to realize that no complete dry-out zone is predicted with this model which means that the flow of liquid water will reach the repository throughout the thermal loading cycle. In a similar thermohydrologic investigation, Ho et al. (1997) also predict that with an infiltration rate of 10 mm per year, no dry-out zone will develop around the repository.

By contrast, when an infiltration rate of 0.1 mm per year was used with the site-scale model (Haukwa et al., 1996), the results revealed that the boiling zone will be extensive and dry-out conditions will develop and last for about 2000 years. Presumably, during the dry-out period, there will be little, if any, water contacting the engineered barrier. Thus, the magnitudes of the infiltration rate and the subsequent percolation flux at the repository level are critical factors in determining when the engineered barrier will first be subjected to degradation from the thermal waters. Assuming that the infiltration rate is of the order of 5 mm per year or more, the fluid system resulting from the boiling conditions described above will be predominantly hot water with whatever gas phase remains from the original ambient conditions in the fractures and matrix.

Effects of Thermal Field on Fracture-Matrix Interactions

One of the most important considerations in the expected behavior of the repository is the fracture-matrix interaction that determines the flow rates in the matrix and the fractures, the resulting convective heat transfer, and the thermochemical and thermomechanical response of the site. The ability to successfully model and assess the thermohydrologic conditions depends to a large degree on the understanding of the flow and displacement in this complex fracture-matrix system. Although fracture characterization in the ESF is currently ongoing, certain features have been identified: (1) nearly straight fracture traces, but with a small tortuosity; (2) some variations in dip, but a large fraction of fractures with nearly vertical dip; (3) widely variable apertures; and (4) some degree of interconnection of fractures leading to a fracture network.

Flow and transport in fractures have been well-studied. Single-phase flow permeability has been studied in terms of the fracture mean aperture, fracture statistical properties, and spatial correlation. Capillary-controlled displacement, such as drainage, in a rough fracture has been modeled in analogy to porous media. Because of the complicated nature of two-dimensional flow in fractures, however, saturations of trapped phases can be considerably higher than in porous media. Discontinuous, episodic flow of liquid in a fracture plane has also been analyzed.

Under unsaturated flow conditions, the movement of water is subject not only to capillary suction (imbibition) from the low permeability matrix, but also to gravitational and viscous forces. At sufficiently low percolation rates, capillarity may overcome the viscous and gravity forces, and liquid water will preferentially reside in the matrix. However, if percolation rates are sufficiently high, the flow of water will occur primarily in the fracture, which has a much higher permeability. The critical flow threshold was found to increase with an increase in the matrix permeability and also to be affected by the presence of pore-lining materials at the fracture-matrix interface, which are in turn affected by the thermochemical response.

The following main classes of chemical processes are likely to occur as a result of the strong thermal disturbances that will occur in the vicinity of the repository:

- dissolution-precipitation of minerals from the liquid phase in response to changing temperatures;
- solid-solid phase transformations; and
- mineral precipitation from an aqueous phase in response to vaporization.

Minerals whose solubilities are strongly temperature-dependent, such as the silica polymorphs and the carbonates, will be most affected. Because many of these reactions involve a change in the volume of the solid phases and the release or uptake of water, they have the potential for altering the porosity, permeability, and moisture budget of large volumes of rock. In addition, precipitation-dissolution at the matrix-fracture interface will affect the fracture-matrix interaction. These changes will, in turn, affect the hydrologic and mechanical properties of the repository block.

Mineral deposition in fractures and matrix pores will become particularly important if thermal loadings are sufficiently high to cause boiling and the generation of heat pipes, as may occur depending on the magnitude of the infiltration rate. The net effect of this process will be not only to plug fluid pathways and divert or pond water, but also to cause a possible throttling, or limiting, flow condition. Water movement between the matrix and fractures may be further affected by changes in the vapor pressure of the pore waters as salinity increases during vaporization. Vitric tuffs are most susceptible to the formation of hydroxyl-bearing clays and zeolites through dehydration reactions because of the instability of glass even at low temperatures (Bish et al., 1995).

With regard to the mechanical effects as the temperature of a fractured rock mass increases significantly, the rock mass will undergo thermally induced displacements and changes in stress that, in addition to the material properties, will depend on: (1) the frequency and orientation of the fractures that are present; and (2) the aperture distribution. As the temperature first changes, both the displacements and stresses rise over a relatively short period of time. Thereafter, the rate of increase will become more gradual, so that over longer periods of time, nearly asymptotic values will be approached.

For given material properties of the rock (linear coefficient of thermal expansion, constant Poisson's ratio, and Young's modulus), the theory of linear thermoelasticity can be used to estimate the displacements and stresses for the rock matrix, in response to the thermal load. However, the presence of the fractures introduces a complication because they represent a discontinuity in the rock system. Depending on the geometry of the fracture network, the overall rock deformations will take place in the normal and/or shear modes. As a result, apertures will generally tend to close as the matrix expands, and the net displacements for the rock mass will be less than what is predicted for an unfractured rock. The corresponding implications on fluid permeability and flow paths could be important. Another mechanical effect in response to the temperature increase is thermal degradation due to microcracking. Such cracking is a function of the confining stress and can occur during the opening of an extensile crack being subjected to changes in compressive stress (Blair, 1994).

The chemical and mechanical effects discussed above are closely coupled to the thermohydrological behavior of the rock system surrounding the repository. Collectively this is referred to as a thermo-hydro-chemical-mechanical (THCM) coupled system. The chemical and mechanical effects driven by the thermal load occur relatively close to the repository; and for the most part, they have been considered to be of minor importance by the project team. As in the case of the effect of the thermal field on the infiltration rate, the Panel believes that the basis for this assumption should be clearly stated in the TSPA-VA, because the mechanical response to fractures, during the heating and cooling phases of the repository, could significantly influence the fluid flow field in the vicinity of the waste containers. The Panel urges that the coupled THCM behavior of the repository be examined in order to correctly plan and interpret measurements from the proposed *in-situ* tests, such as the Drift-Scale Test.

C. Engineered Barriers and Waste Package Performance

Overview

An effective Engineered Barrier System (EBS) and Waste Package (WP) construction can enhance the overall performance of the repository. The goal is to design the EBS and the WP for

- a long isolation period to permit essentially complete decay of many of the radionuclides in the waste, and
- controlled slow release of the remaining radionuclides in the waste to the adjacent geologic formation.

The design and materials of construction of the EBS/WP have a major impact on both of these functions. The EBS and WP also have important functions for the placement, pre-closure, and potential retrieval requirements of the project which must be considered in the design; however, this report focuses on the post-closure, long term performance, and its impact on overall repository performance.

The EBS and WP comprise all of the containment barriers and structure from the fuel and waste form (WF) outward to the surface of the drift wall. In project terminology, the Waste Form Degradation deals with conditions inside the EBS/WP system, and the Near Field Environment deals with conditions outside the EBS/WP system.

The durability of the containment system is controlled by the combination of the environment, design, and the materials of construction. The environment includes the existing geologic formation which determines the initial conditions, e.g. chemical composition of solids, waters and gases, and the amount and distribution of water and temperature. The determination of perturbations to these initial conditions that occur as the result of the construction and operation of the repository is a central task of the design and analysis phase of the project. The design of the overall repository and its components determine the thermal, mechanical and chemical conditions at the repository. The materials of construction are then selected to provide the desired durability under the conditions set by the environment and the design.

Transport into Drift

General description

The following comments represent the current understanding of the Panel members. Further review may provide a basis for clarification or modification of this understanding.

The important factors are: (1) the amount of water entering a drift; (2) the chemical composition of the entering water; (3) the form--drips, seepage, water vapor--of water; and (4) the spatial and temporal distribution of water entry. Conditions will vary around and along a waste package, along a drift, and at different locations within the repository. Some areas will have a steady exposure to high relative humidity and otherwise dry conditions. Other areas will have a high relative humidity along with some bulk water entry that will vary with location and time.

Beyond the region affected by the thermal-mechanical effects of the repository, the water will move through the undisturbed, unsaturated zone to the repository level. Water moves more readily through fractures in the rock than through the rock matrix. Flow along fractures is the primary delivery mode for water to the excavated drifts and outer surface of the EBS. Nonuniform water flow is likely due to: (1) episodic fracture flow; (2) dissolution-precipitation of minerals along fractures; and (3) other mechanical and chemical effects on water transport.

Near the drifts, there will be thermal-mechanical-chemical affects of the repository construction and operation. Heat generated by the waste will flow radially outward from the packages through the EBS to the surroundings, and this heat will drive the water from the surrounding rock. There is the potential for refluxing of these waters above the repository. Soluble species can be picked-up and redeposited elsewhere. The water transport--amount and form--and water chemistry will be affected.

Apparent status

There is a clear recognition and consensus that a description of water transport into the drifts is one of the most important parameters required for modeling the repository. The approach of the TSPA team has been to determine boundary conditions for the most and least amount of water flowing into the drifts. Based on the amount of water entering, various scenarios of water flow have been identified, e.g. water vapor, occasional water droplets, steady flow of water droplets, and episodic water flow. Different areas of the repository experience different water conditions, and a category of water flux, e.g. high, medium and low, can be assigned to any given area. The total volume of water must be conserved, i.e. high flux in an area must be offset by lower flux in other areas. A rationale is required for changing these categories of water flux with time.

The processes which cause changes in the chemical composition of the incoming water due to interactions with the rock surrounding the drifts have been identified by the project. Some models are being developed to deal with these changes; however, there is no working model to determine the water chemistry. Information that defines the range of environmental conditions that may exist for the incoming water is sketchy and requires refinement. Furthermore, there is little or no experimental data to validate the project model or to compare that model's predictions to those from alternative models. No compilation exists of relevant data and experience from natural analogues, industrial experience, or the scientific literature.

Waste Package Performance

Models for EBS and WP Performance

Since the development and documentation of models in this area for the TSPA-VA are in progress, the Panel has not yet performed a thorough review. However, the following comments are based on the Panel's review to date.

Apparent status

No working model or set of models exists for the prediction of performance of the EBS and WP. There are several process models in various states of development that deal with specific aspects of performance, e.g. steel barrier corrosion, corrosion resistant metal corrosion, waste form alteration and radionuclide transport. The goal is to refine and add to models from prior TSPAs to include more of the physical and chemical realities of the processes.

It is a realistic goal to incorporate an ensemble of models for EBS and WP performance that account for the physical and chemical realities of processes that determine performance. It is an unrealistic goal to strive for a set of fully deterministic models for the EBS and WP to support TSPA-VA. Several of the processes are stochastic, e.g. pitting corrosion, and none of the processes can be dealt with totally from first principles. There are spatial and temporal variations in many of the controlling parameters, e.g. temperature, water transport and water chemistry. A probabilistic treatment of these processes can be used to characterize actual behavior.

The project approach to pitting corrosion of a corrosion resistant barrier demonstrates this treatment. Pitting corrosion of the waste canister metals is a primary determinant of performance of the WP. A goal for TSPA-VA is to improve upon the treatment of pitting, and specifically to incorporate more of the physical and chemical affects on WP pitting. There is wide agreement among corrosion scientists and engineers that the environmental factors important to the pitting of passive metals include chloride ion concentration [Cl], level of acidity (pH), oxidizing potential (Eh) and temperature (T). There is also agreement that pit initiation, pit growth, pit arrest, and reinitiation are different stages of the overall process. However, there is no consensus as to the role of these factors at the atomic level, e.g. the chloride ion role in the breakdown and repair of the passive film that protects the metal. The project model for pitting proposes to deal with the effects of Cl, pH, Eh and T on the various stages of pitting, but the project model will not deal with pitting at the atomic level.

In evaluating the performance of the proposed repository, the approach has been to identify all possible degradation modes, to determine the likelihood that any of these modes will affect a waste package and to determine a set of damage functions, e.g. time-to-penetration and penetrated surface area with time. The effects of the damage are then included in the performance models. There is a tendency in this approach to focus on the worst case scenarios. While this may lead to conservative determinations, it must be determined whether it is realistic. The worst conditions for all processes may be mutually exclusive.

There has been a tendency to generalize on the effects of specific processes and hardware on repository performance. In several cases, these generalizations are not backed-up by sufficient technical analysis and data. This can lead to overly optimistic or overly pessimistic conclusions. Examples where the current technical support does not justify the generalized assumptions and/or conclusions include:

- Cathodic protection by the outer barrier will extend the life of the inner barrier.
- Microbially induced corrosion (MIC) will degrade performance of the waste package.
- Ceramic drip shields will improve performance of the waste package.
- Radionuclide transport from a penetrated waste package readily occurs and takes place rapidly.

Any of the assumptions may be correct; however, technical analysis is required to determine the direction and magnitude of the effect on performance.

<u>Paucity of experimental data</u>

In the area of EBS and WP performance, there is little or no project developed data to provide guidance for the evaluation of design alternatives, to guide the development of process models or to test alternative models. There is an equivalent lack of a compiled information base for: (1) analogue models; (2) directly related projects; and (3) relevant data and service experience from relevant applications. Some data sets, presently incorporated in the TSPA-VA, are anecdotal with insufficient technical documentation. The quality assurance requirement on laboratory experimental programs appears to have had a large negative impact on the development of needed experimental data.

The paucity of experimental data is having a major negative impact on the TSPA-VA. The uncertainty is significantly increased and transparency is significantly reduced. To support the development and validation of the EBS and WP process models, experimental data from controlled laboratory experiments, closely monitored field studies, and well documented service applications are quite useful, but these data for repository conditions are scarce. There are well accepted laboratory methods to obtain data for important factors such as corrosion resistance of metals, corrosivity of environments, local water chemistry in pits and crevices, and dissolution-precipitation of minerals. The data gathering and experimental validation of models are more tractable for the in-drift processes affecting the waste packages and waste form (EBS/WP/WF) than for the far-field processes which require evaluation of thermal-mechanical-chemical interactions with the geologic formation.

As will be noted in the discussion that follows, the problems created by the lack of experimental data and, in turn, the inability to validate the various models being developed for analyzing the performance of the proposed repository are generic in nature. These problems apply to essentially all of the analyses related to the engineered barrier system and the waste package, including:

- analyses of the transport of species from packages damaged by water,
- development of models for analyzing/predicting water chemistry, deposits, and corrosion products,
- the mechanisms of corrosion processes, including stress corrosion cracking, and
- development of models for analyzing thermal and radiation-induced embrittlement processes.

A wide range of water chemistries within the drifts has been projected. The repository waters are fairly innocuous to corrosion resistant metals; however, several processes have been identified that can modify these waters and potentially make them more corrosive, e.g. concentration of soluble salts by evaporation and boiling, crevices, and MIC. Current projections of possible water chemistry range from highly acidic to highly alkaline and moderately reducing to strongly oxidizing. Because of this low level of understanding and high level of uncertainty, earlier analyses have assumed that: (1) all metals are potentially exposed to their most corrosive conditions; (2) the probabilities of corrosion initiation and rates of propagation are the highest; and (3) rates of degradation and limits to performance are at their maximum levels. Reducing the uncertainty and increasing the understanding will narrow the range of water chemistries and guide the selection of metals that have increased corrosion resistance under these conditions. Data are needed to move in this direction.

Significant progress needs to be made in the determination of realistic ranges of water chemistries at the waste package and waste form under repository conditions. Progress needs to be made, including the development of experimental data, in gaining a better understanding of the corrosion behavior of waste package metals exposed to realistic water chemistries.

Stochastic waste package degradation model

Many of the processes that determine repository performance are stochastic. The approach is to develop models of these processes that capture this stochastic nature while including important physical and chemical parameters that affect the response. The goal is for physical and chemical parameters to be treated in a manner that will achieve an outcome that is realistic. For waste package degradation, this approach combines process models for important degradation processes, along with the influential environmental conditions, e.g. temperature, relative humidity, time-of-wetness, and water chemistry.

Models for water chemistry, deposits and corrosion products

The determination of the chemical composition of waters entering, residing in, and exiting the EBS and WP is crucial to corrosion, waste form degradation, and radionuclide transport. There is no “standard” model within the project for the treatment of processes that affect the chemical composition of waters, dissolution and deposition of solid species, or formation of corrosion products. Several process models incorporate sub-routines to determine water chemistry and stable phases, while others have place-holders for the required values to be provided from another phase of the project. There is no formal data base or compilation of chemical and physical properties available for these calculations. A centralized data base would benefit both transparency and traceability.

There are important inputs to the corrosion models that are not available or, where available, contain large uncertainties. These major inputs include temperature, time-of-wetness, water chemistry, and form of water. Water chemistry includes pH, chloride ion, oxidizing potential (Eh), and level of aeration. The form of water includes thin condensed films, droplets, water in porous deposits or prior corrosion products, and standing water. There are also spatial and temporal variations.

The water chemistry and stable species can be changed by several possible processes: crevices and restricted geometries, MIC, EBS interactions, formation of salt films and other deposits, and formation of corrosion products. The approach to handle these changes to water chemistry in the corrosion model is to develop a set of sub-routines that will predict the changes that occur to incoming waters as a function of the particular process. For example, the corrosion of iron in a crevice will decrease the oxygen concentration and increase the ferrous ion concentration of water in the crevice. Active MIC can affect the pH and oxygen concentration of the water. Neutral waters flowing through porous concrete will pick up alkalinity. The development of these sub-routines is in progress.

It is not clear how the various sub-routines to deal with crevices, deposits, corrosion products will be turned-on and turned-off over the time period of interest. Will they be used to determine the broadest range of possible conditions? Will they be triggered by other events or inputs?

Models of corrosion resistant metal corrosion-pitting

The treatment of localized corrosion of the corrosion resistant metals (CRM) is the most important issue related to waste package degradation. There is no universally accepted model for dealing with localized corrosion; however, there is broad agreement as to the nature of the process and the controlling parameters of the process. The approach is to develop a stochastic model that incorporates the important physical and chemical parameters of the real process.

The treatment of pitting corrosion uses a set of process parameters to describe the initiation and growth of pits. Important features include pit initiation, growth of stable pits, arrest of pit growth, and reinitiation of arrested pits. A set of probability functions describes the likelihood of each of these events. The probability functions are in turn a function of the corrosivity of the environment. The initiation, growth, and reinitiation process are favored by increased corrosivity, while the arrest of growing pits is diminished by increased corrosivity. A metal that had low initiation, growth, and reinitiation factors along with a high arrest factor would be highly resistant to pitting. Outputs of the model are distribution of pit depths with time, time-to-first penetration. and penetrations with time.

The morphology of the damage as it progresses through a thick-walled barrier has not been addressed. Corrosion science has dealt almost exclusively with pit morphology in thin-walled specimens. The penetrated surface area as a function of time is desired for input to transport calculations for moisture and other species into and out of the packages.

Models of corrosion resistant metal corrosion-internal corrosion after penetration

After penetration of the inner barrier, moisture can enter the package and contact the spent fuel, vitrified waste packages, and other internal components of the canisters. Presumably, the treatment of water chemistry changes and the formation of deposits and corrosion products will be similar to that described for processes outside of the waste canisters. Consideration will need to be given to the interactions among incoming waters and the materials and species present inside the packages. Internal corrosion of the packages will be dealt with as pitting corrosion under the environmental conditions determined by the water chemistry model and deposit and-corrosion product models.

Models of corrosion resistant metal corrosion-galvanic protection

Galvanic protection from localized corrosion has been observed when corrosion resistant metals are coupled to more active metals. Iron in many waters will be more active than a CRM, and credit was taken for protection of the CRM inner barrier by the beneficial galvanic action with the outer steel barrier. While this process is feasible, there has been no model or formal technical treatment developed to examine the effect. Does the beneficial effect arise from a shift in potential of the CRM or from the chemical modifications to the environment that accompany the anodic and cathodic reactions? There are no project experimental data to evaluate the effect or to guide in the development of a model.

Considerations in this area include determination of corrosion potentials of steel and CRM and anodic polarization behavior of steel and cathodic polarization behavior of CRM. The effects of iron oxide corrosion products on the potential and polarization need be considered. It is unlikely that oxide covered steel is a very efficient anode. The geometric effects include corrosion product filled crevices, sizes of areas of exposed CRM and corroded steel, and separation distances between anodic and cathodic sites. The effects of limited moisture and moisture in porous corrosion products will increase ohmic effects. Chemical modifications of the environment in contact with the CRM (cathode) need be considered as well as the shifts in corrosion potential.

Models of carbon steel corrosion

The approach of the TSPA team is to develop process models for dry conditions and wet conditions. A switch is incorporated into the performance model to select wet/dry corrosive environment based on the temperature-relative humidity-surface conditions. Dry corrosion pertains during the extended, hot period after repository closure, and wet corrosion pertains when the packages have cooled sufficiently for the formation of condensed moisture on the metal surface. Wet corrosion rates are much higher than dry corrosion rates.

Wet corrosion of steel is modeled as a general penetration rate with an added pitting factor to account for non-uniform, localized corrosion. Temperature is accounted for in the model. Enhancements to the model are in progress to account for water chemistry (pH and chloride ion) and salt deposits. The effect of prior corrosion products appears not to have been addressed. Credit has been taken for galvanic action benefiting the inner barrier; however, the accompanying detrimental effect on the steel has not been addressed.

As noted above, the experimental data base to support the wet corrosion model development and to validate the model and alternative models is sparse and inadequate.

The effect of thick layers of iron oxide, corrosion products on the outer surface of the package and within the gap between the outer and inner barriers has not been considered. The stresses that develop from the growth of thick layers of iron corrosion products in restricted geometries have resulted in mechanical damage in other applications.

Dry corrosion of steel is modeled as a general penetration rate with an added pitting factor to account for localized corrosion. The penetration rate is a function of time, temperature, relative humidity and sulfur dioxide concentration. Project-developed data are available to support and validate the dry corrosion model.

Microbially induced corrosion

There is no model for dealing with microbially induced corrosion. MIC is not another mode of corrosion degradation, rather MIC acts through the modification of the water chemistry in contact with a metal. The changes to water chemistry can then influence the metal's resistance to the common modes of corrosion, e.g. general corrosion, pitting and crevice corrosion. The project plans to treat MIC as a process that alters the water chemistry and to determine the resulting water chemistries and their effects on the waste package and waste forms. It has been established that there are several organisms capable of participating in MIC that are present in the repository. It has not been established whether there is sufficient water and nutrient to sustain MIC. Active MIC could increase the corrosivity of the waters to carbon steel (the CAM outer barrier). It has not been established whether active MIC would result in water chemistry modifications that are more corrosive than abiotic conditions would be to the corrosion resistant metals (the inner barrier).

Stress corrosion cracking

There is no model for dealing with stress corrosion cracking (SCC). The primary source of tensile stresses is residual stresses in welds. Additional potential sources of stress are other fabrication processes, package placement and post placement, for example, rock falls. The plan is to treat SCC in a manner similar to pitting. Initiation, propagation, and arrest will be considered. There are little or no project data for SCC.

Embrittlement processes-thermal and radiation

There is no model for dealing with these embrittlement processes. There is little or no project data for these thermal and radiation-induced processes.

Mechanical effects on waste package

Rock-fall effects on waste packages have been modeled. Scenarios included were the size and weight of rock masses and the distances of fall. The mechanical effects on packages were determined with and without backfill.

Uncertainties and Complexities in the Analysis of the EBS and WP

The following areas have been identified for further improvement and/or clarification of their treatment in the TSPA-VA process. In several areas, there is no apparent consensus as to how the repository will perform. In other areas, the repository behavior is better defined; however, improvements are required to describe the behavior with models amenable to the integration and abstraction requirements of TSPA-VA.

Refluxing & patterns of water transport in the near field

The amount, form, and chemical composition of water entering the drifts are crucial to understanding and evaluating EBS and WP performance. Spatial and temporal variations are certain to occur.

Determination of water chemistry, corrosion products and deposits

This information is crucial for assessing the corrosion of the waste package, the degradation of waste form, the mobilization of radionuclides, and the interactions with the EBS and near field. The methodologies for these determinations are not well documented and there are no project data for use in these process models.

Localized corrosion of CRM in thin moisture films vs. complete immersion

Corrosion processes in thin-electrolyte films can differ significantly from those in which there is complete immersion in bulk electrolyte. The former is the norm for this application. Questions to be addressed include: Can pitting be sustained through thick-walled structures under these conditions? What is the effect on penetration rate?

Progression of corrosion damage for thick-walled corrosion resistant metals

The shape and distribution of corrosion damage through thick layers of CRM have not been addressed. Corrosion science has dealt almost exclusively with pit morphology in thin-walled specimens. The predictions of time-to-first penetration and the rate of increase of penetrated area will be affected by this in a major way.

Transport into, through, and from the waste package

The transport paths are likely to be lengthy and along restricted passages. Pits in the barriers are likely to be small and filled with corrosion products. Transport will often be along thin layers of moisture.

Internal waste package environment after penetration

The internal environment will control and be modified by interactions with the waste forms. This, in turn, will affect transport and subsequent corrosion of the WP from the inside.

Galvanic effects of outer barrier (CAM) on inner barrier (CRM)

The technical basis for this effect requires further development and clarification. Electrode potentials expected for steel and CRM have not been defined. Steel covered with iron oxide corrosion products is not an efficient cathode. The geometry of the electrochemical cell requires consideration of the area of steel available, the CRM area to be protected, and the current flow through the electrolyte and prior corrosion products.

Thick layers of iron oxide on the corroding outer barrier

There is a significant volume expansion in going from iron to iron oxide. The presence of voluminous deposits of iron oxides on the WP needs to be considered. Chemical/electrochemical effects can arise from the ferric ion in hematite (Fe_2O_3). Mechanical effects can arise from the growth of the iron oxide on the outer surface and between the outer and inner barriers once the outer barrier has been penetrated.

Microbial induced-corrosion

Several organisms capable of participating in MIC are present in the repository. Two crucial issues have not been addressed. First, whether there are sufficient nutrient and water to sustain MIC within the EBS and WP. The form, magnitude, and supply of nutrient is a crucial issue, and this needs to be quantified before an evaluation can be made as to the feasibility of MIC within the proposed repository. Secondly, how will MIC modify the corrosivity of the waters to the CRM? Will the environmental conditions, e.g. pH and Eh, be modified to an extent that a more corrosion resistant metal is required?

Backfill

Backfill in the drift to cover the waste package provides additional mechanical support and distribution of loading in the event of rockfalls. The backfill will also affect the transport of water to the surface of the WP. The backfill will break and disperse “drips” onto the package. If it is fine (sand) then capillary action needs to be considered, and if it is coarse (gravel) then bulk transport is more likely. There can be effects on water chemistry, deposition and corrosion products. The effects have been identified, and their treatment is in the preliminary stages.

Ceramic drip shield or coatings

The incorporation of a ceramic drip shield as part of the EBS is being considered. The concept is that a monolithic ceramic shield above the package will disperse, deflect, or prevent water from dripping directly onto the outer barrier surface. There is no technical analysis to support this concept or to provide a basis for inclusion in an EBS/WP performance model.

After some initial period, the outer surface of the package will be covered with a relatively thick layer of corrosion products and deposits, and water drops will not have direct access to the steel surface. Rather, the water will be transported through the porous corrosion products and deposits to the metal surface. If backfill is included in the EBS, this dispersion of droplets and prevention of direct impingement will also be realized without a drip shield.

Without backfill, it is uncertain that a shield and its support structure could be designed to survive rock falls. The likelihood of condensation on the bottom of the shield with subsequent moisture entrapment and refluxing needs to be considered.

Ceramic coatings on the steel outer barrier of the WP are being considered as a design option. There is no technical analysis to support an assessment of the effects of a ceramic coating on the waste package performance. The ceramic coating provides corrosion protection to steel as a barrier to prevent moisture from contacting the steel. The application of a continuous, ceramic coating that will survive the rigors of handling during emplacement is problematic. No protection is provided at pores, cracks, or other breaks in the coating. As corrosion of the exposed steel proceeds, the volume expansion of corrosion products will cause breaking and spalling of the surrounding coating, and corrosion will proceed over a larger area. The benefit of this coating for waste package performance on the time scale of interest to the repository needs to be considered.

Dual CRM versus CAM/CRM

A design alternative is a dual-walled waste package with both containers being a thick-walled (approximately 2 cm) corrosion resistant metal. This design is more robust with respect to corrosion. Penetration will occur after longer times, and the stochastic nature of localized corrosion will spread the distribution of penetrated packages over longer periods. Once penetrations occur, the localized corrosion sites provide a tortuous path for transport of moisture and species into and from the packages. The chemical and mechanical effects and interactions with thick layers (15 cm) of iron oxide corrosion products are removed from consideration. The apparent beneficial effects in terms of analyzability and uncertainty should be evaluated.

D. Waste Form Degradation and Radionuclide Release

Overview

A wide variety of waste types and compositions are candidates for disposal at the proposed Yucca Mountain Repository. Under current policy 90 percent of the waste (63,000 metric tons) will be spent commercial fuel and the remainder (7,000 metric tons) will be defense waste. Most of the defense waste will be high-level waste that has been vitrified and placed in canisters Additional waste types that are considered for disposal at Yucca Mountain include: Navy reactor fuel, DOE fuel from research reactors, and waste forms (glass, crystalline ceramic, or mixed-oxide fuel) resulting from the immobilization and disposition of excess weapons plutonium. The spent fuel assemblies and the canisters of vitrified waste are to be combined into 11,000 disposal containers.

These waste types vary substantially in their chemical characteristics, radionuclide inventories, radiation fields, and thermal outputs. For the purposes of this interim report, comments are focused on issues related to the spent commercial fuel, as it accounts for a major portion of the radioactivity and contains most of the long-lived actinides. This report, however, does not address the performance of spent fuel cladding. Preliminary comments are made on the vitrified HLRW glass, which contains mainly the heat-generating, short-lived, fission products which remain as waste after reprocessing.

The waste form (spent fuel or vitrified waste) is the first barrier to radionuclide release. Indeed, in their initial configuration in the repository, the radionuclides are entirely confined in the waste forms. To the extent that there is successful containment, the dependence on far-field barriers (e.g., long travel times, dilution in groundwater, or sorption on mineral surfaces) is substantially decreased. More importantly, to the extent that the waste forms are a significant and successful barrier to radionuclide release, uncertainties in the analysis of the performance of the far-field barriers may become less important. Thus, the Panel believes that it is important, as far as possible, to exploit the properties and behavior of the waste forms as important barriers to radionuclide release.

The principal means of release of radionuclides from the waste form is by corrosion in the presence of aqueous solutions. Thus, in the TSPA-95, models have been developed to describe the release of radionuclides due to alteration and corrosion of waste forms in contact with water. The TSPA-95 assumes that certain nuclides, such as ^{14}C, ^{36}Cl and ^{129}I, may be released as gases and escape unimpeded from the EBS (although there is a discussion that the dominant release mechanism for ^{129}I may be via solution). Principal repository parameters affecting release rates are: the timing of the contact of the waste form with water, the nature (vapor, thin-films, flowing) and volume of the water, groundwater compositions (e.g. pH, Eh, chloride content) and temperature. These are time-dependent parameters, that is, their values are very sensitive to the projected or modeled future of the repository. Principal materials parameters affecting release rates are:

- thermodynamic stability of the waste forms;
- the kinetics of alteration and corrosion reactions;

- surface area;
- speciation of released nuclides in solution;
- the solubility limits of potentially precipitated phases; and
- the formation of alteration products.

In general, these material parameters can be measured by laboratory experiments or, in some cases, predicted from fundamental scientific theories or bounded by empirical results.

Even prior to contact with water, the properties of the waste form may be affected by external parameters. Oxidation of the UO_2 can cause volume changes. Thermal events may cause devitrification of the glass; radiation fields can change the stability of crystalline phases (e.g., increase the leach rate) or cause microstructural changes (e.g., radiolytic formation of bubbles in the waste glass). As discussed at the TSPA-VA abstraction meeting on "Waste Form Degradation and Radionuclide Mobilization," there are really two options in modeling the relevant phenomena:

- develop simple, but essentially deterministic, models of waste form degradation and mobilization of radionuclides which include the most important processes to repository performance; or
- develop a "response surface" which is apparently a fit to existing data as a function of relevant parameters generated by thermohydrologic models (e.g., temperature and relative humidity).

The present TSPA approach is a combination of these two methods, both of which are limited by the requirement that they be computationally efficient.

Regardless of the final approach, the results must capture the essential chemistry and physics of the corrosion and mobilization processes, they must be testable (that is confirmed by laboratory experiments or field data), and they must be transparent in terms of the conceptual models that are utilized and traceable in terms of the input data and parameter ranges selected.

In the TSPA-95 analysis, initial bounds on radionuclide concentrations in the groundwater were derived using a waste form dissolution model. These concentration values are then compared with solubility-limited concentrations which are generally sampled from concentration distributions (many determined by expert elicitation). The lowest concentration value is used in subsequent calculations. When solubility-limited values are used, each isotope of the element in question is precipitated in a solid phase according to its isotopic proportions. The entire waste form surface area is assumed to be exposed to the near-field environment as soon as the first pit penetrates the waste package. The waste form is assumed to be covered by a thin film of water and alteration processes are immediately initiated. For the spent nuclear fuel (SNF), a semi-empirical model is used which is expressed as a function of temperature, total carbonate concentration and the pH of the water in contact with the waste form. The analysis did not take credit for any benefit from the cladding or containers (e.g., vitrified waste canisters) located within the waste package. There was also no explicit consideration of transport of radionuclides by colloids.

In the TSPA-95 analysis of potential releases from the WP and the EBS, sensitivity analyses were conducted to identify key parameters that have a large impact on important performance measures such as the peak release rate from the EBS (OCRWMS M&O, 1995. Chapter 8.3.9). The three most important model parameters were: ^{99}Tc solubility, the infiltration rate, and the spent fuel dissolution rate. Release rates increased with higher thermal loads. These results illustrate the dependence of waste form behavior on the modeled repository parameters (i.e., infiltration rate and thermal field and duration) and the importance of the inherent materials properties (e.g., dissolution rate) on the release of radionuclides (e.g., ^{99}Tc).

Waste Form Descriptions

A detailed description of the types and variability of the waste forms is apparently not yet available. The "Preliminary Waste Form Characteristics Report/Version 1.0" (Waste Form Report) (Stout and Leider, 1994) is very much a living document for which many additions will be necessary. For purposes of the preliminary waste form characteristics report, the authors have divided the waste forms into three categories: spent fuel, glass, and other waste forms. The detailed description of the characteristics of the waste forms consists of a compilation of data (mainly tables and figures without text or discussion, but with the original reference indicated). Thus, at this time, the Panel has not been able to evaluate the variability of the waste forms and the associated effects on potential radionuclide releases.

Key issues in the Waste Form Report (e.g., 3.4.1.3: "Solubility Controls on Radionuclide Concentrations in Solution: Preliminary Results for U, Np, Pu and Am") are only briefly discussed (4 pages). The compilation of concentrations in solution is based on calculations using geochemical codes with no appropriate comparison to experimental results. There is no discussion of the thermodynamic data base used in these calculations. The identified secondary phases resulting from experiments bear little relation to the calculated concentrations. The quoted references are surprisingly old (e.g., LLNL draft report, November, 1990). Very little data are cited from the published literature.

Thus, the waste form characteristics report can be considered only as a preliminary version (as the report is labeled) which is essentially not ready for detailed review. However, the detailed characterization of the wastes will be an essential part of the performance assessment analysis.

Alteration and Corrosion Mechanisms for Spent Fuel

Under oxidizing conditions in the presence of water, or even moisture, the UO_2 in spent nuclear fuel is not stable. In oxic solutions, uranium has a strong tendency to exist as U^{6+} in the uranyl molecule, UO_2^{2+}. Uranyl ions react with a wide variety of inorganic and organic anions to form complexes. Throughout most of the natural range of pH, U(VI) forms strong complexes with oxygen-bearing ions like CO_3^{2-}, HCO^{3-}, SO_4^{2-}, PO_4^{3-}, and AsO_4^{3-}, which are present in most oxidized stream and subsurface waters. At 25°C and with a typical groundwater P_{CO_2} of about 10^{-2} atm, the most abundant of these are the uranyl carbonate species, which are stable down to a pH of about 5. Below pH = 5, U(VI) is generally in the form of UO_2^{2+}. Thus, in most near surface environments, uranium is easily transported in natural waters, as the U(VI) uranyl phases have generally high solubilities.

Additionally, the reaction kinetics are rapid. Wronkiewicz et al. (1996) estimated that UO_2 pellets in experiments (90°C) in which water was dripped on to the pellets would have been completely altered in 660 years. They concluded,

> The speed at which the UO2 reactions occurred suggests that an advanced stage of uranyl alteration phase development can develop almost instantaneously within the constraints of the time-scale expected for repository disposal.

These results also imply that alteration may occur simply due to the exposure to moisture in an unsaturated repository setting. It is important to note that a typical rock at Yucca Mountain with a porosity of 11 to 14% and water saturation of 40 to 70% contains more water than low porosity rocks in hydrologically saturated sites (Murphy, 1991). The results concerning the formation of secondary phases (Wronkiewicz et al., 1996) are also supported by kinetic and thermodynamic models of UO_2 corrosion (e.g., Bruno et al., 1996) in which the final steps of the corrosion process are the precipitation and subsequent dissolution of the secondary uranium phases.

Not only are the reaction kinetics for the alteration of UO_2 rapid, but the phases that form will depend on the composition of the reacting groundwater and temperature. Recently, Murphy has pointed out that for certain key alteration phases (e.g., schoepite), the solubility product increases with decreasing temperature. Due to this retrograde solubility, source term concentrations in solution will increase as the temperature of the repository decreases due to radioactive decay and heat dissipation. The general cooling of the repository will also lead to the ingress of water, and the solubility increase will be augmented by the increasing thermodynamic stability of dominant aqueous uranyl carbonate complexes with decreasing temperature (Murphy, in press).

The “Preliminary Waste Form Characteristics Report/Version 1.2” (Stout and Leider, 1996) provides a summary of experimental results obtained within the context of the Yucca Mountain Project (YMP). The evident restriction to YMP data has led the authors to deprive themselves of much of the available and pertinent data in the literature (particularly on alteration phases of UO_2). Still, based on this data base, the authors conclude:

> Thus, the corrosion of spent fuel is more complicated than anticipated. The retention of fission products and actinides cannot be predicted at this time without further examination of additional grains and fragments of reacted fuel, an understanding of the grain boundary penetration and the increase of surface area, and the distribution of radionuclides between reacted phases and solution. While our studies suggest that the alteration phases will incorporate a large proportion of the radionuclides that have been released from dissolved spent fuel and that such a process may act as a significant mechanism for retarding the migration of radionuclides from the waste package, synergistic effects between the waste form and parameters affecting its corrosion, with other components of the repository,

> must be taken into account before using the present data in predicting the fate of radionuclides in a repository.

No references to the studies mentioned were provided in the report.

The authors of the TSPA-95 (6.2-1) use a deterministic conceptual model for the description of the alteration/dissolution of the spent fuel. Incorporated into the model are the following considerations: (1) instantaneous release mainly from the gap and fuel grain boundaries; (2) matrix release due to the dissolution of UO_2; (3) solubility-limited radionuclide concentrations in the aqueous phase. This model appears to capture some of the critical parameters: temperature, pH and total carbonate concentration. However, the ranges for the solubility limits used are mainly based on expert elicitations. No experimental program is proposed by which these elicitations might be confirmed. This dissolution model was criticized in the "Total System Performance Assessment 1995 Audit Review" prepared for the U. S. Nuclear Regulatory Commission (Baca and Brient, 1996, pages 5-6 to 5-8). Their criticisms included:

- The effects of dry oxidation on the surface area of the fuel have been neglected.
- The neglect of colloid formation and transport may be nonconservative.
- There were inconsistencies in the calculated values for leach rate (perhaps due to differences in the assumed pH for each set of calculations).
- It was not clear how the temperature, carbonate and pH values were selected for the calculation of the distribution function for the dissolution rates of spent fuel.

The Panel believes that all of the above points are well justified. The issue of surface area is particularly important, as the dissolution rate scales directly with the surface area. The higher oxides (formed during dry oxidation of UO_2) pass through structural modifications that cause both reductions (a few percent) and expansions (> 10%) in volume. This may result in disaggregation of the fuel with a substantial increase in surface area.

The TSPA-95 may be very nonconservative because it fails to include the role of colloids; however, there is a careful discussion of the potential effects of colloids and how these effects might be incorporated into the next TSPA. The Panel has not yet reviewed the results of such an incorporation.

A "response surface" approach in modeling the spent fuel corrosion is described in the Waste Form Characteristics Report (versions 1.0 and 1.2). This dissolution model is said to be based on concepts from nonequilibrium thermodynamics. The specific connection to the deterministic models used in TSPA-95 is not clear; however, the conclusions are important to note:

> Currently, detailed knowledge is not available for the atomic (mechanistic) steps and the sequence of chemical/electrochemical reaction steps to describe the dissolution process over the range of spent fuel inventory, potential water chemistries, and temperatures.

and

> The development of a release rate model is more complex than a dissolution rate model. The release model includes dissolution rates, precipitation rates, colloidal kinetics and adsorption rates. At this time the approach is semi-empirical and depends strongly on the unsaturated testing experiments to provide data and chemical process models.

This is a rather demanding set of requirements for a “release rate model” that is both semi-empirical and based on nonequilibrium thermodynamics. Also, the Panel is not convinced that the “unsaturated testing experiments” can provide all the data required to develop the corrosion model.

The description of the derivation of the response surfaces begins from first principles with equations that are said to represent the correlation between corrosion and the electrochemical rate data. A single data set was used to determine dissolution rates by regression analysis which used a 21-term quadratic polynomial of five variables. As an example of the approach:

> A third-order term with burnup, oxygen concentration and inverse temperature was included to better represent the apparent effects of radiolysis. The equation with the smallest root-mean-square error and largest correlation coefficient (r2 = 0.91) was a 13 term model.

It is too early to comment on the scientific basis for this approach, but there appear to be a few deficiencies:

- Because this approach is not based on a deterministic model of the relevant processes, it is not clear how such an approach can be “tested” or “challenged”.
- The regression analysis is simply a fit to a limited data set. Therefore, this approach is already an “abstraction.” The dissolution models being used to describe spent fuel corrosion certainly are not detailed process models. This approach stands in contrast to efforts in Europe that are focused on developing mechanistic models for cladding failure, instantaneous release from the gap, release from grain boundaries, and finally, matrix dissolution of the UO_2.
- There is no apparent effort to test the response surface model by comparison to other experiments.
- There is no justification given for the extrapolation of the results to conditions outside those of the original data set or over longer periods of time.
- The recent Nuclear Waste Technical Review Board (NWTRB) report (NWTRB, 1977) emphasizes that the first desirable characteristic of a TSPA is “transparency”. These dissolution models are not transparent. Further, they are cast in a form that is fundamentally untestable.

At present the following parameters are expected to affect spent fuel corrosion rates and the resulting formation of alteration products and the related release of radionuclides into solution: Eh, effective surface area (geometric vs. grain boundary), pH, solution compositions, solubility-limiting phases, colloid formation, and radiation effects on alteration phases and colloids. These parameters are not discussed in terms of a deterministic model which can then be used for a subsequent abstraction. There is also no discussion of the status of the data base to model the reaction progress and predict the stable, radionuclide-bearing phases or the kinetics of reactions related to the spent fuel corrosion. The detailed analysis and evaluation of DOE models will require a discussion of these controlling parameters. At the moment there are only limited data available for analysis; and in the absence of appropriate experimental programs, there can be no quantitative analysis or confirmation of the conceptual models.

In summary, it is not at present possible to explain short term experimental results, particularly the distribution of radionuclides between the solid phases and solution. This would, perhaps, not be such a serious situation if the dissolution rates were low or if the corrosion mechanisms were well understood and described by deterministic models.

Uncertainties and Complexities in the Analysis of Spent Fuel Corrosion Models

The following areas were identified for further improvement and/or clarification of the treatment in the TSPA process. The Panel emphasizes that further discussions with DOE scientists and contractors may clarify certain issues; however, it is already evident that some of these issues require a considerable amount of additional work (as outlined in some cases by the Abstraction/Testing plans).

- The data base used to develop the deterministic models of spent fuel corrosion is too restricted to be useful or convincing. Considerable data are available in the published literature that could be used to amplify present deterministic models or test calculated "response surfaces."

- Changes in solution composition will have an important effect on dissolution rates of the spent fuel, vitrified high-level waste, and canister materials as a function of time, flow rate, and temperature. It is not evident to the Panel how these analyses will be done.

- Only limited use has been made of natural analogues. This is a serious omission since the basis for the use of natural uraninite, UO_{2+x}, as a structural and chemical analogue for the long term behavior of UO_2 in spent nuclear fuel is well documented.

- Release rates are very sensitive to the surface area of the spent fuel exposed to water. What is the relationship between the corrosion and alteration of the spent fuel and the increase in surface area with time? What is the effect of the formation of precipitated alteration phases?

- Both the spent fuel (by oxidation) and vitrified waste (due to temperature and radiation field effects) will be "conditioned" prior to contact with water. What will be the physical and chemical state of the waste forms at the time that they come in contact with water?

- The use of solubility-limits to determine radionuclide concentrations in solution can only be done with respect to specific solid phases. To what extent are these phases identified? Are processes such as co-precipitation being considered?

- Will the expert elicitations used to define the solubility-limited concentrations be replaced by laboratory determinations of these values?

- The fundamental thermodynamic data base on actinide-bearing phases, which may form as corrosion products or as solubility-limited precipitates, is very incomplete. What is the status of efforts to improve this data base?

- The secondary, alteration phases are mentioned in many contexts (e.g., host phase for certain radionuclides, solubility-limiting phases), and potentially may cause cladding failure due to volume expansion. A number of statements and assumptions are made concerning their formation (e.g., ". . . the precipitation rate and the quantity of precipitates are directly proportional to the intrinsic dissolution rate"). How important will the behavior of these phases be in the overall performance of the near-field barriers? What experimental programs are in progress to confirm the properties and behaviors of these phases? What are the effects of temperature and radiation on the properties of these secondary phases?

- Colloid formation and transport must be included in future performance assessment analyses.

- Can any credit be taken for waste form containers (e.g., DWPF canister)?

- The consideration of fuel cladding as a potential barrier is relatively recent in the TSPA, and preliminary models were used in the TSPA-95. What is the relationship between spent fuel corrosion (e.g., formation of alteration products) and cladding failure?

The Panel notes that many of the above issues were discussed, or at least noted, during the abstraction workshops. In fact, the list of issues related to waste form degradation and radionuclide migration are impressively complete. Some issues, however, were not discussed in detail (e.g., evolution of solution compositions within the waste package container, the use of natural analogues for spent fuel corrosion, and radiation effects on hydrated secondary phases).

Alteration and Corrosion Mechanisms for Borosilicate Glass

Although the total fraction of activity in the borosilicate glass "logs" is low (7% of anticipated repository inventory) and many of the radionuclides are short-lived fission products, the importance of the release of this fraction of the activity should be carefully analyzed. As an example, one of the options being considered for disposition of excess weapons plutonium is to mix the Pu in dilute concentrations in the glass "logs" which contain defense-generated high level waste.

At the time of this report, the Panel has not examined in detail the data base, corrosion models, or the abstractions of the models used to describe radionuclide releases from the glass "logs". A discussion of this subject will be included in a subsequent Panel report.

However, several issues may be raised:

- What is the rationale for mixing spent fuel assemblies with glass logs in the same waste package container? This has the effect of increasing the temperature and radiation field for the glass logs. The reactivity of glass is much increased at elevated temperature; the effects of temperature and an ionizing radiation field on the microstructure of the glass cannot be neglected.

- Has the potential for and effects of reactions between spent fuel assemblies and the vitrified waste been considered? Could these two waste types be more easily and simply disposed of in separate parts of the repository?

- Both waste forms, spent fuel and borosilicate glass, benefit from a dry environment due to the absence of water. However, does this requirement for a "dry" environment (e.g., maintaining elevated temperatures by the heat from radioactive decay) lead to deleterious or difficult to predict behavior for the waste forms?

Transport of Species from Damaged Waste Packages

The TSPA-95 treatment of the transport of radionuclides from the waste package appears to be ultra-conservative and describes release rates considerably higher than is realistically expected. Although the current models provide outputs for time-to-penetration of the inner barrier and the number of penetrations with time, there is a need for a better description of the progression of damage after the inner barrier has been penetrated. Parameters such as the geometry of pits, distribution of pits along and around the barriers, and the penetrated surface area are of interest. The presence of corrosion products and deposits within the pits will affect the transport of moisture, as well as other parameters that influence radionuclide releases. These factors are important to the determination of transport of radionuclides from the waste package. This transport is likely to be by arduous and tortuous paths.

There is little or no experimental data for the transport of radionuclides under conditions that pertain to damaged waste packages. The transport of moisture and soluble species are of interest. Bench top and full scale tests to monitor entry, distribution within and egress from a waste package with pitting penetrations would aid in reducing the uncertainty.

Summary Observations and Conceptual Issues

The Panel is not aware of any general conceptual description of the physical and chemical state of the near-field environment over time as a function of repository parameters (e.g. flow rate, physical and chemical effects of backfill, thermal regime, etc.). This is particularly important in analyzing the long-term behavior of the canister and waste form. In the absence of such a description, there is an unfortunate tendency to use unrealistic bounding assumptions (e.g., the canister is considered to have failed completely when there are a certain number of corrosion pits that breach the canister, or the UO_2 in spent fuel is assumed to have corroded completely with the first appearance of water in the canister). In contrast, it is more likely that even in the presence of extensive pitting, the amount of water entering the canister will be small. This water may not escape the canister either due to the lack of an accessible flow path or due to reaction with the spent fuel. The alteration products of UO_2 commonly have significant molecular water (e.g., 12 molecules of water for each formula unit of schoepite, a common U^{6+} alteration product). The alteration products of UO_2 occupy considerably larger volumes than the UO_2; thus, precipitation of these alteration products may "clog" the flow paths. Estimates of the total volume of water are essential to determining whether concentrations in solution will be limited by precipitation of solid phases. On the other hand, the alteration products are quite sensitive to temperature (losing water at elevated temperatures and undergoing structural modifications), and some of these phases show retrograde solubilities. Alteration phases may become host phases for some radionuclides (particularly the actinides) and not for others, such as Tc-99.

Thus, the "picture" of the near-field environment as a function of time must be an essential part of the TSPA-VA. At the moment, a clear statement of the physical and chemical environment in the near-field is not part of the TSPA, although physical elements of this description are included in TSPA-95, Chapter 6.5.5 "Advective Release from Waste Package and EBS" (CRWMS M&O, 1995). Additionally, alternative engineered barrier system conceptual models which include a consideration of diffusive and advective release for pitted canisters are considered in Chapter 8 Section 8.3.5 of the TSPA-95 (CRWMS M&O, 1995). The authors conclude,

> The implication is that the 'partially failed' waste containers by pitting corrosion should still be able to perform as a potentially important barrier to radionuclide release, and EBS transport models that incorporate more realism should be considered in future EBS performance analyses.

This issue is all the more important in light of the recent recommendation by the NWTRB (NWTRB, 1997, page xiii),

> Given the inevitable uncertainties about repository performance, more attention to defense-in-depth (multiple, redundant barriers) is needed in the waste package and repository designs. In particular, *comprehensive* studies of alternative engineered barriers -- such as fillers, backfill materials, drip shields, and engineered inverts -- should be completed.

E. Disruptive Events and Criticality

In later reports, the Panel plans to discuss the effects on the proposed repository of earthquakes, volcanism, criticality, and human intrusion. Because of the early stage of the Panel's review, the only such topic that is treated in this initial report is criticality. Even in this case, the ensuing discussion will be limited.

Criticality

Overview

The attempt to address the issue of criticality at Yucca Mountain through performance assessment is still in its early stages; it was not addressed systematically in TSPA-95 or earlier TSPAs, and so TSPA-VA will be the first attempt at integrating criticality analysis with the larger PA model.

The problem is that a very large number of critical masses, of both plutonium and uranium-235, will be emplaced in the waste canisters, and although this material will be in configurations that initially will be designed to preclude criticality, there is the possibility that a critical mass could be reassembled later in time as the engineered barrier features degrade.

The task of TSPA-VA in this arena would ideally be to perform a set of realistic analyses of various potential criticality scenarios, if feasible, or to produce bounding analyses if such would be adequate for the purposes of the overall TSPA-VA project.

Regulating against criticality

Another important piece of background concerns the relevant regulation to be used in judging criticality. Currently, USNRC has a regulation, 10 CFR 60.131 (b)(7), adopted many years ago for governing criticality in deep-geological repositories like the one proposed for Yucca Mountain. This regulation reads as if it is necessary to preclude a criticality with very high confidence. Unfortunately, the regulation does not clearly indicate whether it is intended for the operational phase (pre-closure), the post-closure phase, or both.[2]

It seems clear to the Panel that the USNRC's intent when this part of 10 CFR 60 was written was to regulate operational-phase criticality, in which case strong language seeking very high assurance is fully appropriate, and in the Panel’s view easily achievable. However, the ambiguity in the applicability of the regulation leaves the assessment of the proposed repository up-in-the-air. The Panel believes it vital that the USNRC clarify the applicability of its current criticality regulation as soon as possible; and if pre-closure criticality alone is the intent of the current regulation, the USNRC should initiate actions to determine how to regulate post-closure criticality sensibly. (For the reasons cited below. the Panel believes that the current criticality regulation, if applied to post-closure time period, is not a sensible approach.)

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[2] The USNRC’s current criticality regulation is at Title 10, CFR, Part 60, “Disposal of High-Level Radioactive Wastes in Geologic Repositories.” Part 60.131 is entitled “General design criteria for the geologic repository operations area.” Part 60.131(b)(7), entitled “Criticality control,” reads:

> (7) *Criticality control.* All systems for processing, transporting, handling, storage, retrieval, emplacement, and isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. Each system shall be designed for criticality safety under normal and accident conditions. The calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a 5% margin, after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation.

The TSPA-VA team's current approach

For TSPA-VA, it is intended that the criticality analyses will be done separately, as a set of "off-to-the-side analyses" in parallel with the mainline analysis of future repository performance. That is, criticality will not be incorporated into the mainline models for TSPA-VA, but will be analyzed separately.

The current approach is to divide the criticality analysis into three physically distinct problems: (1) in-package criticality (after degradation begins of the integrity of the packages or of their contents); (2) near-field, in-the-drift criticality after material migrates out of the canisters into the drift space; and (3) far-field criticality, defined as anywhere outside the drift. For each of these three regions, the TSPA-VA team is seeking to develop realistic models of the behavior of the materials that could affect criticality, and then to develop a workable analysis, aimed at quantifying (if feasible) both the likelihood that a critical configuration could occur in the distant future, and the consequences, meaning the processes that might produce larger peak radiation doses than would otherwise be the case.

Accomplishing this set of analyses fully is recognized by everyone involved to be a very ambitious task, perhaps not achievable in this TSPA-VA round. Even replacing the word "fully" by the word "adequately" in the previous sentence may not work --- there may be more unknowns in the problem than can be resolved, or bounded, in the time remaining for the completion of TSPA-VA. This seems to have led the team to concentrate its efforts on two types of work in parallel: first, to try to develop the best analyses they can, and second to develop alternative approaches in case -- which is a likely possibility -- there is the need for fall-back analyses of certain key topics.

Panel evaluation of the approach

The Panel believes that the two key elements of the approach above -- allowing criticality to be studied through side analyses instead of in the mainline TSPA modeling, and developing fall-back strategies in parallel with working on the mainline criticality analyses -- are both sensible.

The Panel concurs with the notion that the difficulties involved in a realistic analysis of the many possible criticality phenomena may be too large to produce a strongly supported set of models and results. The problem, in a nutshell, is that whether a critical configuration can occur, and the consequences thereof, depends on the detailed assumptions regarding various future phenomena, such as differential chemical dissolution/leaching rates as a function of chemical conditions, differential canister failure scenarios involving specific types of canister corrosion failures, moderator effectiveness over time, negative or possibly positive feedback mechanisms, and so on.

The Panel believes that, depending on what figure-of-merit is used in the ultimate regulations for the proposed repository, it may be that whether the facility "passes" or "fails" will depend on the details of the regulations and their method of implementation to a much larger extent than for any other of the important phenomena that may occur at Yucca Mountain in the future. Specifically, the Panel believes that:

- If the regulations require that analyses "preclude" criticality from ever occurring within the proposed repository for all future times, or for any regulatory time period beyond when canister failure begins, demonstration of compliance may not be possible.

- If the regulations allow criticality to occur but place a probabilistic bound on its occurrence, then the Panel believes that it may be feasible to demonstrate some bound on the likelihood, provided the likelihood that would still be acceptable is not extremely low.

- If the regulations treat criticality just like any other event or process, and only require that the ultimate health-effect criterion (say, public dose or risk) be met, then in the Panel's view it ought to be feasible to perform a set of criticality analyses (either reasonably realistic analyses or bounding-type analyses) to ascertain how criticality phenomena contribute to the ability of a repository at Yucca Mountain to comply with regulatory requirements.

The Panel’s judgment on the above three points is based on the following (preliminary) observations:

- Despite all of the best efforts that the criticality modelers will bring to bear on the subject, it is the Panel’s judgment that it will probably not be possible to preclude criticality processes with high confidence over the full future time covered in the TSPA. This is likely the case even if only a 10,000-year regulatory period is to be covered, and all the more true if much longer times, such as a million-year horizon, require study. This is because the details of the ways that the canisters may fail, and the ways that materials may chemically interact and move (both in-canister and in-drift), may not be knowable in sufficient detail.

- Nevertheless, it seems to the Panel likely that analysts will be able to place probabilistic bounds on the likelihood of a criticality occurring in-canister and also in-drift, and that such bounds will likely provide very useful insights into the role of criticality in the overall performance of the proposed repository at Yucca Mountain.

- More importantly, the Panel believes it likely that the consequences of any of the criticality events being considered, were they to occur either in-canister or in-drift, can be understood or at least bounded well enough to allow for a sensible regulatory review.

Therefore, the Panel believes that those responsible for the current TSPA-VA criticality work need to emphasize developing enough information about the potential consequences of various criticality scenarios to derive useful values for, or at least bounds on, these consequences. This emphasis need not detract from the emphasis of the current work that is seeking to model the events and processes that might produce a criticality; it would be an additional emphasis that the Panel believes is quite important at this early stage of wrestling with criticality in the proposed repository at Yucca Mountain.

F. Transport

The TSPA-VA evaluates transport, or movement, of radionuclides from the waste to points where humans may be exposed. As discussed here, the analysis is based on processes that have been studied and modeled at many sites, but where the heterogeneity of the Yucca Mountain site leads to uncertainties in both competing process models and in parameters. The transport process from the repository begins with the release of radionuclides from the waste form and their incorporation into a passing stream of thermal water, either by dissolution or in the form of colloidal aggregates. The colloidal aggregates will occur in the natural system and may serve as sorptive sites for radionuclides. In addition, colloids from degradation of the waste forms and repository materials may be introduced into the SZ if they remain in the contaminated groundwater as it passes through the UZ.

In addition to the water-based transport described in this section, gaseous radionuclides such as carbon-14 (in the chemical form of CO_2) may be released to the atmosphere. The TSPA-VA assessment of gas pathways and releases is not reviewed in this interim report.

Transport in the UZ

The migration of fluids in the unsaturated zone down to the water table will be subjected to the effects of the various factors discussed in Section II.A, such as lateral diversion, that complicate the geometric path of migration. Radioactive waste heat and the introduction of man-made materials into the repository could influence the geochemistry of the fluids entering the drifts via seepage, as well as the fluids in which the radionuclides are traveling. In addition to the corrosion rate of the waste package and the dissolution rate of the waste form, the chemistry of the fluid seeping into the drift will affect the solubility of key radionuclides. The chemistry of the fluid transporting the radionuclides will influence their speciation and sorption.

As in the flow above the repository, an important consideration in the UZ process is the fracture-matrix interaction that determines the rate of migration for the thermal waters in the regions below the repository. Molecular transport of dissolved radionuclides is by diffusion into the matrix and by advection in the fractures. The transport of colloidal aggregates, on the other hand, occurs mostly by flow in the fractures.

Retardation

Diffusion of solutes from fractures into the matrix of the volcanic tuffs is expected to contribute significantly to the effective retardation of radionuclide movement in both the UZ and SZ. Numerical simulations indicate approximately complete diffusive exchange between the fractures and matrix for travel times in excess of ~100 years using representative material properties (Robinson, 1994). Sorption of radionuclides may also play an important role in retardation and the attenuation of peak concentration. Sorption can slow transport of some radionuclides by orders of magnitude and is therefore an important effect. Recent tracer test results at the C-wells and from laboratory studies suggest that sorption both in the matrix and on fracture surfaces (Triay et al., 1996) is a significant and verifiable process. Obviously, the relative transport rates will be significantly affected by the distribution of the percolating liquid between the matrix and the fractures.

Transport in the SZ

Transport in the SZ occurs by similar mechanisms, and flow in the fracture is dominant. Results of studies of dispersion in a fractured system have been reported in the literature, using both continuum and discrete models. The amount of dilution by dispersion is directly related to the spatial variability of hydraulic conductivity. Its precise characterization has a direct bearing on the appropriateness of the conceptual models for flow and transport (e.g., continuum vs. discrete flow; dual-porosity vs. effective porosity transport). Even if one assumes a Fickian model of dispersion, there exists considerable uncertainty in parameter values for longitudinal and transverse dispersivity. Recent tracer tests from the C-well complex at YM indicate effective longitudinal dispersivity of about 4 to 50 m for travel distances of 30 to 100 m. An analysis of theoretical macrodispersivity based on air permeability data from the TSw indicates an average value of about 14 m for the longitudinal dispersivity (Altman et al., 1996), and tracer test results from the Paleozoic carbonate aquifer yield an estimate of 15 m for this parameter over a travel distance of about 120 m (Claassen and Cords, 1975). Realizing that dispersivity can be scale dependent (it depends on the correlation structure of the permeability heterogeneity), the field scale measurements from YM, when projected to travel distances of 5 to 30 km, indicate dispersivity values for transport on the order of 100 to 500 m.

Transport of colloidal aggregates

Transport of radionuclides in the form of colloidal aggregates bypasses processes, such as matrix diffusion and sorption, that retard the movement of radionuclides. On the other hand, filtration of colloids in geometric constrictions in the fractures, is a process with significant attenuation potential. Colloids are expected to exist in the UZ environment. These colloids may be generated by the undisturbed system (natural colloids), by the radioactive waste (waste-form colloids), or by other introduced materials in the repository (introduced colloids). Radionuclides may be adsorbed by these colloids or be part of the colloid structure (waste-form colloids). Measurements of natural colloids in water samples from the saturated zone suggest that their concentrations are too small to have much effect on radionuclide transport (CRWMS M&O, 1995; Ogard, 1987).

Little is known about the possible magnitude of UZ radionuclide transport due to introduced or waste-form colloids. Pu and Am released from waste glass are expected to exist largely as a waste-form colloid (Bates, et al., 1992). Colloids are expected to be generated from concrete and iron used in the proposed repository (CRWMS M&O, 1995). However, the quantitative characteristics and transport behavior of these colloids were not described in TSPA-95. Work is underway to provide a more detailed treatment of colloids in TSPA-VA.

G. Biosphere, Doses, and Health Risks

Estimating the doses and associated risks to the exposed population will be the final step in assessing whether the proposed repository will comply with the applicable standards and regulations. While the form of the applicable standards is not yet known, certain features can be anticipated. Based on the requirements of Section 801 of the Energy Policy Act of 1992 and on the findings of the National Research Council report, *Technical Bases for Yucca Mountain Standards* (National Research Council, 1995a), it appears likely that the primary basis for evaluation of the performance of the proposed repository will be the doses or risks to a specified hypothetical individual or critical group. The basic approach taken in TSPA-95 to deal with these uncertainties was to make calculations for a variety of possible regulatory endpoints. In conducting the analyses, both individual doses and cumulative releases within 10,000 years, and individual doses at later times, out to 1,000,000 years, were considered.

Identification of Pathways

Comparison of the potential repository performance with the standards will likely require estimating the probabilistic distribution of doses to a critical group as well as conversion of the resulting doses to the associated health effects and risks. Calculating the probabilistic distribution of doses requires identifying the potential pathways of radionuclides from the repository to the biosphere, which comprises the air, water, food, and other components of the landscape that are accessible to humans, as well as the humans themselves; projecting the concentrations that will be present in air, water, food, and other materials with which the exposed population groups might come into contact; and estimating the probabilities that humans will be exposed to contaminated air, water, food, or other materials leading to a radiation dose.

Estimations of radionuclide releases that can lead to contamination of the various environmental systems/pathways cited above will be based on the assumed behavior and performance of the various components within the repository system. These include the canisters or other waste forms in the repository horizon; the backfill, disturbed rock and other materials of the near-field zone in the vicinity of the waste; the rock, air, and water of the unsaturated zone (rock and pores above the water table); the local atmosphere above Yucca Mountain; the water table aquifer immediately beneath the repository; the aquifer downgradient of the repository (away from the repository along the direction of groundwater flow) from which water may be withdrawn via wells for human use; and the regional discharge zone of the groundwater flow system where water exits the ground as discharge to surface water bodies or through evapotranspiration.

Releases to the atmosphere directly above or adjacent to Yucca Mountain can cause exposure by inhalation to the people who might be present in the immediate vicinity and who constitute a potential critical group. Exploitation of groundwater by some potential critical group downgradient from the repository can lead to exposure via food, water, or other contact with contaminated water, including the use of the water for irrigation. If radionuclides are transported through the groundwater flow system to the regional discharge area, they or their radioactive decay products, as well as accompanying toxic chemicals originating in the repository, might accumulate in soil, water, and/or plants, leading to possible exposures to yet another potential critical group.

Calculation of Doses and Health Risks

Assuming that the important pathways that could lead from the repository to human exposure have been identified, the technical feasibility of developing performance assessment calculations to evaluate compliance with a risk standard depends on the ability of risk analysts to model the relevant processes that lead to radionuclide transport, taking into account the dilution or concentration of the radionuclides along these pathways. The adequacy of the performance assessment also depends on the feasibility of quantifying the probabilities associated with any processes that cannot be predicted in a purely deterministic manner.

Further complicating this process is that each of the conceptual, mathematical and numerical models for evaluating these processes has a multitude of uncertainties. These relate to estimates of the doses to the exposed population groups, as well as calculations for converting these estimates into the associated risks. With respect to estimating the doses, the uncertainties relate to those associated with the models for assessing biosphere transport of various radionuclides, with the values of the key parameters that serve as input to those models, and with failures to account for variabilities in these parameters (Hoffman and Miller, 1983). With respect to estimating the resulting risks, the key uncertainties relate to those in the epidemiological studies that serve as a basis for developing the risk coefficients, uncertainties in the doses received by the exposed populations on which these studies are based, and uncertainties in the transfer of these observations from a population in one country (in this case, Japan) to another (National Research Council, 1995b).

Reduction of Uncertainties

Obviously, every effort should be made to reduce these uncertainties. In terms of the uncertainties associated with the risk coefficients, it is important that the analysts keep abreast of related studies underway within the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP). In terms of the uncertainties associated with the dose estimates, one of the first steps is to identify the key input parameters. One method for accomplishing this, as well as identifying the important contaminants and exposure pathways, is through the use of preliminary screening calculations (NCRP, 1996b). Screening calculations can also be used to provide an estimate of upper and lower bounds on doses to the exposed individuals. Once the key input parameters have been identified, related uncertainties can be reduced through gathering additional site-specific data, taking care to direct primary attention to those components or parameters that dominate the uncertainties. The gathering of such data should continue until the uncertainties are either acceptable for decision making or, because of limitations either of time or financial resources, they are irreducible. Under these conditions, decisions must be made in the face of uncertainties that, for practical purposes, cannot be reduced (NCRP, 1996a).

When relevant site-specific data are not available, the most defensible method for obtaining subjective probability distributions is through the formal elicitation of expert judgment. Formal elicitation can be used to encode what is known and not known about an uncertain model component. Expert elicitation then focuses on refining the quantification of the state of knowledge for these components. In the absence of formal expert judgment, important assessments should be assigned to at least two independent organizations, and the assessment should include resources for resolving discrepancies (NCRP, 1996a). In any event, it is important to reduce such uncertainties wherever possible. In certain cases, it may prove more cost-effective to reduce the uncertainties in some of the key biosphere pathways than to reduce the uncertainties in the engineered barriers or geosphere components (Smith et al., 1996).

The Panel believes that certain aspects of the biosphere analyses may be specified when the EPA completes the development of its standards. Until such time as the standards are issued, the Panel encourages members of the TSPA-VA team to meet with appropriate EPA representatives to discuss these matters and to obtain any guidance that may be available. Such guidance, even if only of a general nature, could lead to considerable savings in time and effort. One example is whether DOE needs to be concerned with the dose conversion factors and their associated uncertainties, discussed above, or will these be provided by EPA. Another relates to conditions within the biosphere at times far into the future. In this regard, it is important for members of the TSPA team to know whether EPA plans to define a reference biosphere for purposes of conducting the performance assessment. Similar comments could be made with respect to the exposure scenarios that are to be used in the analyses. For these and other reasons, the Panel urges that DOE authorize the TSPA team to initiate discussions with EPA at the earliest possible date.

III. INITIAL FINDINGS REGARDING TSPA-VA DIRECTION AND PROGRESS

A. Overall Conceptual Model of Repository Performance

There appears to be a lack of an overall description or conceptual model of how the repository is expected to perform during future time periods. The Panel recommends that such a conceptual model of repository performance be described in an introductory chapter of the TSPA-VA. In particular, there is a need to list future possible events, the time periods and sequence in which they may occur, and how they will effect one another. This description of repository performance over time should include:

- conditions in the unsaturated zone above the repository;
- composition of the waste, taking decay into account;
- duration of the hot, dry period, and associated humidity conditions;
- changes in the liner and in-drift chemistry;
- chemical conditions and vapor/liquid water reaching waste packages during and after the hot, dry period;
- corrosion penetration rates and progression of damage through the outer corrosion allowance material;
- corrosion penetration rates and progression of damage through the inner corrosion resistant material;
- physical and chemical processes in the canisters that lead to degradation of spent fuel and vitrified waste;
- rates of mobilization of radionuclides from the waste packages;
- processes and rates of retardation of radionuclides in the near-field;
- travel times through the unsaturated zone, considering alternative flow paths, retardation, and breakthrough;
- dilution and movement of radionuclides in the saturated zone;
- behavior and movement of released radionuclides within the biosphere and avenues of exposure and uptake by nearby population groups;
- assumed locations and lifestyles of nearby populations; and
- assumed values for various factors used to translate radionuclide concentrations into dose and risk estimates.

There is at present no general conceptual description of the physical and chemical state of the near-field environment over time as a function of repository parameters (e.g. flow rate, physical and chemical effects of backfill, thermal regime, etc.). This is particularly important in analyzing the long-term behavior of the canister and waste form.

B. Hydrologic Issues

The percolation flux is one of the most critical parameters, both for interpreting current conditions at the site and in assessing its suitability as a potential repository. Data from several different field observations suggest that the flux reaching the repository level by migration through the matrix is relatively small, which implies that fracture flow is the dominant factor.

A large hydraulic gradient to the north of the potential repository has been identified as a key issue. Because this gradient is located upgradient from the repository, its presence may or may not be relevant to the performance of the repository. It is difficult, however, to argue that an adequate understanding of processes in the SZ exists without considering plausible alternative conceptual models of this feature.

C. Thermal, Chemical, and Mechanical Issues, Excluding Engineered Barriers

The thermomechanical response could be an important factor affecting fracture characteristics and permeability, depending on the degree to which the fractured rock mass undergoes thermally induced displacements and changes in stress with a consequent change in permeability. These coupled effects, driven by the thermal load, will occur relatively close to the repository; and for the most part, they have been considered by the project to be of minor importance. It is the Panel's view that the basis for this assumption should be clearly stated in the TSPA-VA. The Panel is concerned that the mechanical response of fractures during the heating and cooling phases of the repository could significantly influence the fluid flow field in the vicinity of the waste containers.

The Panel believes that the coupled (thermo-hydro-chemical-mechanical) behavior of the repository needs to be examined in order to correctly plan and interpret measurements from the proposed *in-situ* tests, such as the Drift-Scale Test. The present analysis does not adequately address the mechanical and chemical aspects of this coupled behavior.

D. Engineered Barrier System and Waste Package Performance

No working model or set of models exists for the prediction of the performance of the EBS and WP. There are several process models in various stages of development that deal with specific aspects of performance, e.g. steel barrier corrosion, corrosion resistant metal corrosion, waste form alteration and radionuclide transport. The goal should be to refine and add to the models from prior TSPAs to include more of the physical and chemical realities of these processes.

It is a realistic goal to incorporate an ensemble of models for EBS and WP performance that account for the physical and chemical realities of processes that determine performance. Because several of the processes are stochastic, e.g. pitting corrosion, and none of the processes can be dealt with totally from first principles, it is unrealistic to seek a set of fully deterministic models for the EBS and WP to support the TSPA-VA. A probabilistic approach is required. [Note: the terms "deterministic" and "probabilistic" are defined on page 6.]

There has been a tendency by the project to generalize the effects of specific processes and hardware on repository performance. In several cases, these generalizations are not backed-up by sufficient technical analysis and data. This can lead to overly optimistic or overly pessimistic conclusions. Some examples where the current technical information does not appear to provide adequate support are:

- Cathodic protection by the outer barrier will extend the life of the inner barrier;
- Microbially induced corrosion will degrade performance of the waste package;
- Ceramic drip shields will improve performance of the waste package;
- Radionuclide transport from a penetrated waste package is easy and rapid; and
- Radionuclides will be retained in secondary alteration phases.

Any of these impressions may be correct; however, technical analyses and/or experiments are required to determine the direction and magnitude of the effects on performance.

Significant progress needs to be made in the determination of realistic ranges of water chemistries in contact with the waste package and waste form under repository conditions. Similar progress needs to be made in the determination of the corrosion behavior of waste package metals and waste forms exposed to realistic water chemistries. Experimental data will be needed for these determinations.

E. Experimental Data

In the area of EBS and WP performance, there is little or no project-specific data to provide guidance for the evaluation of design alternatives, to guide the development of process models, or to test alternative models. There is an equivalent lack of a compiled information base to serve as input to the analogue models, and also a lack of relevant data derived from directly related projects. Some data sets are anecdotal, with insufficient technical documentation.

The paucity of experimental data has a major negative impact on the TSPA-VA. The uncertainty is significantly increased and transparency is significantly reduced. To support the development and validation of the EBS, WP, and WF process models, data from controlled laboratory experiments, closely monitored field studies, and well documented service applications are quite useful; however, data of this nature that are representative of repository conditions are scarce. There are well accepted laboratory methods to determine data for important factors such as corrosion resistance of metals, corrosivity of environments, local water chemistry in pits and crevices, and dissolution-precipitation of minerals.

F. Analyzability

The data gathering and experimental validation of models are more tractable for the in-drift processes affecting the waste packages and waste form (EBS/WP/WF) than for the far-field processes which require evaluation of thermal-mechanical-chemical interactions with the geologic formation.

G. Transport of Species from Damaged Waste Packages

The TSPA-95 treatment of the transport of radionuclides from the waste package appears to be ultra-conservative which leads to high estimated release rates. A better description of the progression of damage after the inner barrier has been penetrated is needed. The presence of corrosion products and deposits within the pits will affect the transport of moisture and other products. These factors are important to the determination of the transport of radionuclides from the waste package. The transport of radionuclides is likely to be by arduous and tortuous paths.

There are little or no experimental data for the transport of species under conditions that pertain to damaged waste packages. The transport of moisture and soluble species are of interest. Bench top and full scale tests to monitor entry, distribution within and egress from a waste package with pitting penetrations would aid in reducing uncertainty.

H. Sensitivity Analysis of Thermal Loading Design Concept

Both waste forms, spent fuel and borosilicate glass, benefit from a dry environment, but the requirement for a “dry” environment (e.g., maintaining elevated temperatures due to heat from spent fuel disposal) may lead to deleterious effects associated with high temperatures. In our review to date, the trade-offs are not clear between the benefits from maintaining a dry condition versus the potential for hot conditions to have an adverse effect. In addition, the high temperatures may create conditions that are difficult to predict. The Panel is interested in learning of the degree to which hot conditions increase the complexity of the repository performance assessment.

I. Waste Form Performance

The present approach is a combination of two approaches to model the relevant phenomena: (1) development of simple, but essentially deterministic, models of waste form degradation and mobilization of radionuclides which include the most important processes to repository performance, and (2) development of a “response surface” which is a fit of existing data as a function of relevant parameters generated by thermohydrologic models (e.g., temperature and relative humidity). Both methods are limited by the requirement that they be computationally efficient. Regardless of the final approach, the results must capture the essential chemistry and physics of the corrosion and mobilization processes, they must be testable (that is confirmed by laboratory experiments or field data), and they must be transparent in terms of the conceptual models that are utilized and traceable in terms of the input data and parameter ranges selected.

It is too early to comment with any authority on the scientific basis for these approaches, but the Panel’s initial reactions to the assessment of waste form performance are:

- The data used to develop the deterministic models of spent fuel corrosion are too restricted to be useful or convincing. Considerable data are available in the literature that could be used to amplify present deterministic models or test calculated “response surfaces.”

- Changes in solution composition will have an important effect on dissolution rates of the spent fuel, vitrified high-level waste, and canister materials as a function of time, flow rate, and temperature. It is not evident to the Panel how this analysis will be done.

- Only limited use has been made of natural analogues. This is a serious omission since the basis for the use of natural uraninite as a structural and chemical analogue for the long term behavior of UO_2 in spent nuclear fuel is now well documented.

When the Panel considers the full array of models that will be required to describe canister failure, cladding failure, waste form dissolution, near-field radionuclide release, and additionally considers the sensitivity of these models to important repository parameters (e.g., time of contact with water, volume of water, and thermal history), it is impressed by the complexity of the resulting combination of models. This complexity raises the following questions:

- What efforts are now underway to present the combined models for the near-field behavior of the repository in a transparent and traceable manner?

- Release rates are very sensitive to the surface area of the spent fuel exposed to water. What is the relationship between the corrosion and alteration of the spent fuel and the increase in surface area with time? What is the effect of the formation of precipitated alteration phases?

- The use of solubility-limits to determine radionuclide concentrations in solution can only be done with respect to specific solid phases. To what extent are these phases identified? Are processes such as co-precipitation being considered?

- Are laboratory experiments being conducted to provide data to replace the expert elicitations being used to define the solubility-limited concentrations?

- The fundamental thermodynamic data base on actinide-bearing phases, which may form as corrosion products or as solubility-limited precipitates, is very incomplete. What is the status of efforts to improve this data base?

J. Criticality

The Panel believes it is vital that the USNRC clarify the applicability of its current criticality regulation as soon as possible; and if pre-closure criticality alone is the intent of the current regulation, that the USNRC should initiate actions to determine how to regulate post-closure criticality sensibly.

The current TSPA-VA approach to analyze criticality is to study it through side analyses instead of in the mainline TSPA modeling, that is, the criticality analysis is not integrated into the overall TSPA probabilistic model. In addition, the project is developing fall-back strategies in parallel with working on the mainline criticality analyses. The Panel believes that these two key elements are both sensible.

The Panel concurs with the notion that the difficulties involved in a realistic analysis of the many possible criticality phenomena may be too large to produce a strongly supported set of models and results. Whether a critical configuration can occur, and the consequences thereof, depend on the detailed assumptions regarding various future phenomena.

The Panel’s preliminary judgment is that despite all of the best efforts to model criticality, it is likely that it will not be possible to preclude criticality processes with high confidence over the full future time covered in the TSPA. Nonetheless, the Panel believes that analysts will probably be able to place probabilistic bounds on the likelihood of a criticality occurring in-canister and also in-drift, and that such bounds will likely provide useful insights into the potential for criticalities to occur. Furthermore, the Panel believes that the consequences of any of the criticality events being considered can be understood or at least bounded well enough to allow for a sensible regulatory review.

The Panel believes that staff members responsible for the current TSPA-VA criticality analysis need to emphasize developing enough information about the potential consequences of various criticality scenarios to derive useful values for, or at least bounds on, these consequences. This emphasis need not detract from the current work that is trying to model the events and processes that might produce a criticality; it would be an additional emphasis.

K. Colloids

Transport of radionuclides in the form of colloidal aggregates bypasses processes, such as matrix diffusion and sorption, that retard the movement of radionuclides. On the other hand, filtration of colloids in fractures is a process that can result in significant retardation of radionuclide transport due to colloids. The key radionuclide in this process is plutonium, a major radionuclide in the waste, which does not transport readily as a solute. For this reason, colloidal transport may be the primary method by which this radionuclide migrates. Because colloids were not analyzed in TSPA-95 and because the TSPA-VA work on colloids is in progress and does not exist in a written, documented form at the present time, the Panel has not reviewed work on this issue in detail.

L. Biosphere, Doses, and Health Risks

Although standards have not yet been issued by EPA for a repository at Yucca Mountain, certain features of the standards may be anticipated. Until such time as the standards are issued, the Panel encourages members of the TSPA-VA team to meet with appropriate EPA representatives to discuss these matters and to obtain any guidance that may be available. Such guidance, even if only of a general nature, could lead to considerable savings in time and effort. One issue that may be clarified even before issuance of the EPA proposed standards is whether EPA will specify within the standards certain aspects of the biosphere analyses.

M. Input from Outside Experts

The Panel recognizes that scientists from outside the project are being invited to participate in various elicitation workshops. However, these workshops do not provide for external review of specific technical elements of the TSPA-VA. The project could benefit from the early inclusion of such scientists and engineers, both because the project may learn of relevant studies conducted elsewhere and because the project will benefit from being scrutinized by a broader audience of scientists and engineers.

The abstraction meetings that have just concluded provided more of an overview of the project's technical approach to various issues than do the elicitation workshops. However, the participants in the abstraction process are essentially those contractors and scientists who have previous involvement in DOE nuclear waste management projects. Representatives from outside the project were invited to the abstraction meetings, for example, from the Nuclear Waste Technical Review Board or from the USNRC or the USNRC's Center for Nuclear Waste Regulatory Analyses, but were not permitted to offer comments at the abstraction meeting. The Panel recognizes that this restriction was based on the limited time and full agendas of the meetings. However, an opportunity was missed for the project's technical staff to benefit from timely input and informal discussions with outside experts.

Appendix A: QUESTIONS THAT THE PANEL PLANS TO ADDRESS

In the course of its deliberations, the TSPA Peer Review Panel has identified a range of questions that it plans to address as it proceeds with the review of the TSPA-VA. With the thought that these questions might be of interest to the DOE and contractor staff, they are listed below.

1. Would it be beneficial to conduct a simple screening scope Performance Assessment of the proposed repository? This would provide information on whether its performance, assessed on a conservative basis, appears to be acceptable and, if not, it would help identify the principal sources and/or pathways of exposure and key data needs.

2. What is the process for incorporating new ideas and/or concepts into the repository design? How is feedback from the TSPA-VA factored into the design?

3. In conducting the abstraction process, how many models and/or runs are performed before the decision is made that the problem being addressed is of second order importance? What is the process for terminating such efforts? Without such a process, continuation of efforts on a particular process could lead to a waste of valuable time and resources.

4. How clearly are the assumptions underlying and incorporated into each model delineated and/or specified?

5. What is the process for separating (discretizing) the processes into manageable groups? How is this accomplished spatially and temporally? How is the decision made as to how many groupings each input parameter should be divided into?

6. Depending on the technical complexity of the issue, the analysis of the proposed repository's performance may be done on either a reasonable/representative basis or as a conservative or bounding analysis. Given that the degree of conservatism in the analysis is uneven, how can one distinguish actual sensitivities from analytical artifacts?

7. How will the analysts factor into their assessments the lack of applicable EPA standards and a revised USNRC Title 10, CFR, Part 60? A significant component of this question may be how the EPA requirements on groundwater protection will be handled. What is the project doing with regard to this issue?

8. Where site-specific data are lacking, who coordinates the search for data from non-DOE sources? Who confirms that all relevant data sources have been adequately checked?

9. What is the basis for the lack of incorporation of mechanical uncertainties into the thermohydraulic models used in performance assessment?

10. How are the results from the planned drift tests to be used in TSPA-VA? Are the tests designed to provide data that will be needed or to confirm conceptual models? How do site heterogeneities limit the general applicability of the measured results?

11. The current plan calls for an E-W tunnel to be drilled. What information relevant to TSPA-VA will be obtained from such a tunnel?

12. What is the experimental basis for assumptions regarding canister corrosion?

13. To what degree have efforts to have a repository design that will meet regulatory limits led to design decisions that may be of questionable practicality? That is to say, who makes the judgment on whether the suggested approach represents something that can be manufactured and incorporated into the design? Note: This question relates to #2 above.

14. Has a formal process been established to assure that all groups, particularly those responsible for modeling the various processes and events, understand the overall goals of the TSPA-VA and how the results of their work will fit into the final product? This process should include a clearly defined mechanism through which the insights of the modeling efforts can be used to improve the design of the repository. Note: This relates to questions #2 and #13.

15. Regarding waste loading plans and assessment of repository performance, what is the rationale for mixing DOE-owned spent fuel assemblies with glass logs in the same waste package container?

16. Has the TSPA-VA team considered the potential for and effect of reactions between spent fuel assemblies and the vitrified waste? Could these two waste types be more easily and simply disposed of in separate parts of the repository?

Appendix B: Review Plan

Peer Review Plan for
the Performance Assessment Peer Review

[The approved Peer Review Plan includes a signature page signed by Stephen J. Brocoum and Chris G. Whipple]

Statement of Work

The objective of the Performance Assessment Peer Review is to provide a formal, independent evaluation and critique of the Total System Performance Assessment-Viability Assessment (TSPA-VA) for the Civilian Radioactive Waste Management System Management and Operating contractor (M&O).

Scope of the Review

The TSPA-VA will be conducted by the M&O for the U.S. Department of Energy (DOE) Yucca Mountain Site Characterization Office (YMSCO). The Peer Review Panel members will conduct a phased review over a two-year period to observe the development and completion of the TSPA-VA. The comments, concerns, conclusions, and recommendations of the final Peer Review Report will be provided to the M&O to support the development and conduct of the License Application TSPA (TSPA-LA). As such, the Panel members will consider not only the analytical approach of the TSPA-VA, but also its traceability and transparency.

The Panel members will evaluate the TSPA-VA for its analytical approach, including:

- physical events and processes considered in the analyses,
- use of appropriate and relevant data
- assumptions made
- abstraction of process models into the total system models
- application of accepted analytical methods
- treatment of uncertainties

These aspects will be evaluated within the context of their significance to the long-term performance of a repository at Yucca Mountain.

The traceability of the TSPA-VA refers to the extent to which a complete and unambiguous record exists of decisions and assumptions, and of models and data, and their use in arriving at the results of the TSPA. Traceability is achieved through the documentation and explanation of all decisions made during the analyses.

Transparency means clear and logical documentation. A transparent TSPA will be clear not only to technical analysts, but also to other informed readers. An informed reader is one with an appropriate background in particular aspects such as the fundamental scientific and engineering principles, numerical analytical methods, or regulatory implications.

Quality Assurance

This peer review will be conducted in accordance with the requirements of DOE Administrative Procedure (QAP) 2.5, Peer Review. The review will also be consistent with the guidance provided in the U.S. Nuclear Regulatory Commission's Generic Technical Position on Peer Review for High-Level Nuclear Waste Repositories (NUREG-1297).

Peer Review Group

The Peer Review Group consists of the six members of the Peer Review Panel, project personnel, and *ad hoc* consultants to the Panel.

Peer Review Panel

Six areas of technical expertise have been identified as necessary for a comprehensive review of the TSPA-VA and six Panel members have been selected to provide this expertise. In addition, one of the Panel members will serve as Chairperson. The expertise required and selected Panel members are:

- Chairperson and Risk Assessment - Dr. Chris G. Whipple
- Physics and Nuclear Safety - Dr. Robert J. Budnitz
- Chemistry and Geochemistry - Dr. Rodney C. Ewing
- Biosphere and Health Physics - Dr. Dade W. Moeller
- Materials Science and Metallurgy - Dr. Joe H. Payer
- Hydrology and Fluid Flow - Dr. Paul A. Witherspoon

A complete identification of the Panel members is provided in Appendix C. *[Note: In the actual Peer Review Plan, the Panel is identified in Appendix A.]*

Project Personnel

Responsible Individual - Dr. Stephen J. Brocoum, YMSCO Assistant Manager for Licensing, has functional responsibility for the TSPA that is the subject of this review, per QAP-2.5.

DOE Observer - Mr. Eric T. Smistad, YMSCO Performance Assessment Technical Lead, will serve as observer and lead point-of-contact for interactions between the DOE and the M&O and Peer Review Panel members.

Responsible Manager - Dr. Jean L. Younker, M&O Regulatory Operations Manager, is responsible for the conduct and completion of the TSPA-VA, per QAP-2.5.

Peer Review Coordinator - Mr. Robert C. Murray, Senior Project Scientist, M&O Technical Evaluation Team, will coordinate interactions between the M&O and the Peer Review Panel members.
[Since the Peer Review Plan was written, Thomas Rodgers has replaced Robert Murray as Peer Review Coordinator]

Technical Secretary - Ms. Susan D. Wiltshire, M&O Planning and Communications Consultant, will facilitate interactions among the members of the Peer Review Panel and its consultants. She will have primary responsibility for documenting the activities of the Panel members.

Ad Hoc Consultants

The Panel Chairperson, with support from the M&O, may engage *ad hoc* consultants to provide technical advice to the Panel members on specific aspects of the review. These consultants may serve the Panel members where technical issues arise beyond the established expertise of the Panel members or where the volume of material to be reviewed requires technical support. The consultants will not have permanent standing on the Panel and will report their comments, concerns, conclusions, and recommendations through the appropriate Panel members. However, the consultants will meet the same quality assurance requirements as the Panel members. The Panel members will retain responsibility for assimilating all consultant work into the reports.

Comments, Concerns, Conclusions, and Recommendations

Members of the Panel will work interactively with the M&O and with all others who can contribute to the Panel's work. Panel members can request any information they deem necessary to the success of the review. Panel members will attend project meetings, technical exchanges, and workshops, as needed. They solicit advice concerning materials relevant to the review and welcome input from any group or individual.

Meetings and Interactions

The first and final meeting of each phase of the review (see Review Schedule Section) will be open. The final meeting of each phase will coincide with the first meeting of the succeeding phase in order to minimize the travel burden. Interactions at these meetings will be generally confined to the Panel members and those directly addressing them. The primary purpose of these meetings is to serve the needs of the Panel in gathering and interpreting information, in discussing various aspects of the TSPA as a group, and in planning future work. Time will be scheduled at each open meeting for the Panel to receive comments and questions from the public.

All other functions of the Panel, collectively and individually, will be treated as closed working sessions. The Panel members' geographic locations, other professional responsibilities, and the nature of the review itself dictate that they will conduct their review according to their own schedules. When multiple Panel members meet or interact among themselves about substantive topics, a meeting summary will be prepared and entered into the record of the review. When individual Panel members attend external meetings or engage in interactions or exchanges of information, the Panel members will prepare a written summary of the event and the information exchanged. These records will be entered into the permanent record of the review at the conclusion of each phase.

Reporting

At the conclusion of each of the first three phases, the Panel will prepare and deliver to the M&O an interim letter report presenting their comments, concerns, conclusions, and recommendations. The final phase of the review will conclude with the submittal of the final Peer Review Report. Each Panel member will contribute to the preparation of these reports. In addition, the Chairperson will prepare an executive summary highlighting key issues and noting the general progress of the review. The Panel members will present their findings to the M&O at the final meeting of each phase.

The Panel will report its comments, concerns, conclusions, and recommendations as a body. The Panel members will pursue agreement in their findings, but will not expect consensus among the diverse technical aspects of the review. Reports will indicate where consensus does not exist and individual Panel members may include dissenting opinions, if they so choose.

The M&O will prepare letter responses to each of the three interim letter reports and a final response document for the Peer Review Report. These response will document the disposition of each of the comments, concerns, conclusions, and recommendations made by the Panel members in their reports.

Review Schedule

The Performance Assessment Peer Review will take approximately two years to complete. The Panel selection process began in October 1996, with the solicitation of nominations from the national and international technical communities. Panel selection was completed in January 1997. The review will formally commence in February 1997, and is expected to be completed by March 1999. The review will be conducted in four discrete phases:

Phase 1: TSPA Orientation - February to June 1997

The Panel members will be introduced to the high-level radioactive waste program, the YMSCO, and the performance assessment program. The Panel members will also familiarize themselves with the previous iterations of the TSPA (i.e., TSPA-91, -93, and -95) and related documents. The Panel members will consider the overall scope and approach planned for the TSPA-VA. Individual Panel members, by areas of expertise, will attend abstraction workshops and other pertinent PA interactions, and have access to all information necessary to initiate their review.

The Panel will conclude this phase with their first interim letter report on June 20, 1997.

Phase 2: Modeling, Scenarios, and Abstractions - June to December 1997

The Panel members will follow and review the process modeling, scenario development, and abstractions at a level of detail sufficient to permit them to engage in the subsequent review phases. Interactions between the Panel members, the M&O PA organization, and other sources of expertise and information will continue.

The Panel will conclude this phase with their second interim letter report on December 15, 1997.

Phase 3: Draft TSPA Review - December 1997 to June 1998

The Panel members will observe and review the process of developing the draft TSPA-VA. They will review the draft and final documentation of abstraction activities, and the preliminary and final results of the TSPA-VA during this phase. Their comments, concerns, conclusions, and recommendations will be documented in the third interim letter report.

The Panel will conclude this phase with their third interim letter report on June 12, 1998.

Phase 4: Final TSPA Peer Review - September 1998 to March 1999

Panel members will be provided with copies of the final TSPA-VA report, once it has been completed, for their review. The Panel's comments, concerns, conclusions, and recommendations will be documented in the Performance Assessment Peer Review Report.

The Panel will conclude this phase with their final Peer Review Report on March 1, 1999.

Review Criteria

Among the potential review criteria enumerated in QAP-2.5 (Section 6.3.e), the following are most appropriate to this review:

1. Validity of basic assumptions

2. Alternate interpretations

3. Appropriateness and limitations of methods and implementing documents

4. Accuracy of calculations

5. Validity of conclusions

6. Uncertainty of results

Appendix C: Peer Review Panel

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Acronyms and Abbreviations

atm	atmosphere (pressure unit equal to 1.013 x 10^6 dyne/cm^2 or 14.7 psi)
CAM	corrosion allowance material
CFR	Code of Federal Regulations
Cl	Chloride
CFu	Crater Flat undifferentiated units
CHn	Calico Hill nonwelded tuff layer
cm	centimeter
CRM	corrosion resistant metal
DOE	U.S. Department of Energy
EBS	Engineered Barrier System
Eh	oxidizing potential
EPA	U.S. Environmental Protection Agency
ESP	Exploratory Studies Facility
HLRW	high-level radioactive waste
ICRP	International Commission on Radiological Protection
LBNL	Lawrence Berkeley National Laboratory
CRWMS	Civilian Radioactive Waste Management System
m	meter
M&O	Management and Operating Contractor
MIC	microbially induced corrosion
mm	millimeter
NCRP	National Council on Radiation Protection and Measurements
NWTRB	Nuclear Waste Technical Review Board
pdf	probability density function
pH	measure of the hydrogen ion concentration or level of acidity
PTn	Paintbrush nonwelded tuff layer
SCC	stress corrosion cracking
SNF	spent nuclear fuel
SZ	saturated zone
T	temperature
TCw	Tiva Canyon welded tuff layer
THCM	thermo-hydro-chemical-mecanical
TSPA	Total System Performance Assessment
TSPA-95	TSPA completed in 1995
TSPA-VA	TSPA supporting the Viability Assessment
TSw	Topopah Spring welded tuff layer
USGS	U. S. Geologic Survey
USNRC	U.S. Nuclear Regulatory Commission
UZ	unsaturated zone
VA	Viability Assessment
WF	waste form
WP	waste package
YM	Yucca Mountain
YMP	Yucca Mountain Project

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