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Risk Information to Support Prioritization of Performance Assessment Models

Prepared for:
U.S. Department of Energy
Yucca Mountain Site Characterization Office
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Under Contract Number
DE-AC08-01RW12101

DISCLAIMER

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CHANGE HISTORY

Revision Number	Interim Change No.	Effective Date	Description of Change
0	0	06/06/2002	Initial issue.
0	1	07/30/2002	Addresses comments to increase editorial consistency with other Project documents. No substantive changes are made to the sensitivity studies. Figure legends and notes are modified for clarity. Discussion of plutonium-240 is added to provide more complete description of key radionuclides. Figures and text describing radionuclide contributions to drip shield and waste package neutralization results are added to enhance explanation of physical basis for results. Discussion of waste package performance sensitivity study is clarified. A table is added to show the correlation between TSPA model areas and components with model validation areas. Organization of Section 5 is modified to provide consistency with the reorganization of TSPA model components.
1	0	08/05/2002	Addresses DOE comments to increase consistency of terminology related to regulatory requirements.
1	1	08/29/2002	Modifies change history to describe development of REV 00 ICN 01 of the document. Corrects Figure 16(b). Corrects Table 4. Corrects typographical errors.

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FIGURES

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ACRONYMS

BDCF	biosphere dose conversion factor
CSNF	commercial spent nuclear fuel
DOE	U.S. Department of Energy
HLW	high-level waste
KTI	key technical issue
NRC	U.S. Nuclear Regulatory Commission
SCC	stress corrosion cracking
SSPA	Supplemental Science and Performance Analyses
TH	thermal-hydrologic
TSPA	total system performance assessment
TSPAI	total system performance assessment and integration
TSPA-SR	total system performance assessment for the site recommendation
USFIC	unsaturated and saturated flow under isothermal conditions

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1. INTRODUCTION

1.1 OVERVIEW

This report provides risk information to support risk-informed, performance-based prioritization of performance assessment models to address model validation and U.S. Nuclear Regulatory Commission (NRC) key technical issue (KTI) agreements. More generally, this information will help in the development of the postclosure safety case for a license application. Although the safety case will address the full suite of site and engineered barrier system characteristics that were considered in the Yucca Mountain site-suitability evaluation (DOE 2002), it will place highest priority on those that are important to meeting the postclosure performance objectives required for a license application and to the assessment of expected risk within the context of those performance objectives.

The risk-informed, performance-based approach has not been comprehensively utilized in previous U.S. Department of Energy (DOE) considerations. Consequently, the considerations for validation of performance assessment models and for addressing the KTI agreements have not yet included a comprehensive risk-informed, performance-based evaluation. The principal limitation in this regard has been the need for adequate development of the performance assessment model components. However, these model components have now reached a sufficient level of maturity to support major decisions, including the site suitability decision. Accordingly, these model components are now believed to be sufficiently mature to support such a risk-informed, performance-based approach.

The information provided here is of two types. First, the report summarizes the general understanding of waste isolation that has emerged from the total system performance assessment (TSPA) analyses conducted to support the site suitability evaluation (DOE 2002). This summary identifies the fundamental characteristics of the repository system that would contribute to meeting the regulatory postclosure performance objectives. Second, TSPA sensitivity studies are used to confirm the general understanding of the waste isolation characteristics of the system and to provide quantitative estimates of the relative importance of these characteristics in meeting those objectives. These studies provide a risk-informed, performance-based perspective on the level of model validation needed for the TSPA model components and on the KTI agreements associated with the technical basis for these model components.

1.2 BASIS FOR THE EVALUATION

The evaluation of the TSPA model components provided in this report focuses on the relative importance of these components to expected risk, as expressed in terms of the postclosure performance measures of 10 CFR Part 63. In particular, the focus is on the performance measures associated with the individual protection and groundwater protection performance objectives in 10 CFR 63.113(b) and (c). The measures associated with these performance objectives are (1) annual dose from all radionuclides to the reasonably maximally exposed individual in the first 10,000 years after permanent closure (10 CFR 63.113(b)) and (2) levels of radioactivity of groundwater in the accessible environment in the first 10,000 years (10 CFR 63.113(c)).

For this evaluation, the measure of expected risk associated with these performance measures is the consequence (e.g., annual dose) multiplied by the probability of that consequence. This approach is consistent with the definition of performance assessment in 10 CFR 63.2.

As an example of the appoach, expected risk expressed in terms of the mean annual dose is estimated from the probability-weighted average of the distribution of projected annual doses. A distribution of projected annual doses arises in part because of uncertainty in the TSPA model components. The mean annual dose is estimated by including the full range of possibilities for these components, weighting them by their probability of occurrence and summing over those possibilities. A distribution of projected annual doses also arises because of the possibility of disruptive events. Probabilistic methods are used to simulate possible futures that include both the high-probability expected conditions and the low-probability disruptive events, each weighted by their probability of occurrence. For the first performance measure, each possible future behavior of the repository system is represented by a curve describing the annual dose to the reasonably maximally exposed individual as a function of time. The mean annual dose curve is estimated by calculating each of these annual-dose-versus-time curves, weighting them by their probability of occurrence, and summing the results.

This evaluation does not consider other measures of risk. For example, the analyses could have considered the mean annual dose of a particular scenario conditioned upon the occurrence of that scenario (e.g., annual dose absent any consideration of probability weighting), or they could have considered the most likely consequence (e.g., the peak of the probability distribution for estimated mean annual dose rather than the expected value). However, neither of these approaches provides a means for assessing the role of TSPA model components in meeting the requirements for individual and groundwater protection that have been established by the NRC. As such, they would not provide a means to assess the potential significance of the various TSPA model components. Therefore, it is the probability-weighted estimate of the performance measures that is the focus of this evaluation.

The estimates for the performance measures provided in this report do not provide all of the information needed for decisions regarding the degree of model validation nor the information needed to address KTI agreements. The focus of these decisions is whether the case presented by the DOE is adequate to provide reasonable expectation that public health and safety would be protected and whether the technical basis for that case is adequate. This standard of reasonable expectation that public health and safety would be protected involves additional considerations, including the roles of the natural barriers and engineered barrier system in enhancing resiliency of the system. Such additional considerations would be taken into account in decisions regarding the technical basis for the performance assessments. Accordingly, the information provided in this report does not determine these decisions. The information provided here only serves to inform those decisions.

2. WASTE ISOLATION CHARACTERISTICS OF A YUCCA MOUNTAIN REPOSITORY SYSTEM

The ability of a repository system at Yucca Mountain to meet the postclosure performance objectives of 10 CFR Part 63.113 depends on its waste isolation characteristics, i.e., the ability of the natural and engineered barriers of the system to limit migration of radionuclides to the accessible environment and to members of the public. Analyses leading up to the Yucca Mountain site recommendation and the TSPA analyses conducted specifically for the site recommendation (CRWMS M&O 2000a, DOE 2002) have resulted in a coherent and consistent picture of the waste isolation capabilities of a repository system at the Yucca Mountain site. This section summarizes that picture, both in terms of physical arguments and the results of the TSPA analyses. The consistency between the physical arguments and TSPA quantitative estimates provides confidence in the relative maturity of the TSPA model and the appropriateness of studies using the TSPA model to inform the prioritization of model validation and NRC KTI resolution.

2.1 CHARACTERISTICS AFFECTING INDIVIDUAL PROTECTION

The postclosure performance objective for individual protection is that migration of radionuclides to the location of a reasonably maximally exposed individual be sufficiently inhibited that radiological exposures would not exceed the requirements specified in 10 CFR 63.113(b). These requirements include a limit to the annual dose of 15 mrem. The waste isolation requirements in this case therefore focus on the ability of a Yucca Mountain repository system to inhibit migration of the radionuclides such that the mean annual dose in the first 10,000 years is less than 15 mrem.

Most of the radionuclides that would be emplaced in the repository are relatively immobile in geologic systems. Tables 6-25, 6-26, and 6-27 of the total system performance assessment-site recommendation (TSPA-SR) model report (CRWMS M&O 2000b, pp. 228 to 230) indicate that the total inventory of radionuclides emplaced in the repository would be on the order 10^{10} curies. Strontium-90 and cesium-137 would compose more than 90 percent of that total and neptunium-237, americium-241, and plutonium isotopes would dominate the remainder. These radionuclides are relatively immobile and only a small fraction of the total inventory would be able to migrate to the location of the reasonably maximally exposed individual in 10,000 years. The radionuclide inventory would also contain a small fraction (less than 0.01 percent of the total inventory at emplacement) that might migrate more readily.

After 10,000 years, the strontium-90, cesium-137, and americium-241 would have decayed away, reducing the total radionuclide inventory by about a factor of 300. Plutonium-239, plutonium-240, and neptunium-237 would compose more than 99.98 percent of the total curie inventory at 10,000 years, and the more mobile radionuclides would compose less than 0.02 percent of the total inventory at that time. The issue for individual protection then is how the natural and engineered barriers would limit migration of these various radionuclides to provide reasonable expectation the individual protection limit would be met.

The natural barriers at Yucca Mountain would limit migration in two ways. First, these barriers include the overlying rock of Yucca Mountain that would limit the amount of water that might contact the waste deep underground and mobilize radionuclides. Second, the unsaturated rock below the repository and the saturated rocks below the water table inhibit the movement of the radionuclides that might be mobilized. As a result, only a small fraction of the total radionuclide inventory could migrate to the location of the reasonably maximally exposed individual. Analyses show that the natural barriers alone are sufficient to reduce the potential release of radionuclides by more than 6 orders of magnitude (CRWMS M&O 2000c, Section 3.1.1 p. 3-3).

The engineered barriers also limit migration of the radionuclides. If the waste packages remain intact for 10,000 years, no radionuclides would be exposed to water that might seep into the emplacement drifts or be able to escape from the repository. The individual protection limit would therefore be met categorically. Some fraction of the waste packages could fail early (e.g., due to fabrication defects). However, in this case the drip shields over those waste packages would serve to divert seepage away from the waste. Under expected conditions, therefore, the system of multiple natural and engineered barriers would limit exposure of the waste to water and migration of radionuclides away from the repository. Depending upon the effectiveness of these barriers, the repository system of barriers would provide reasonable expectation that a reasonably maximally exposed individual would be protected.

The individual protection performance objective, however, applies to potential disruptive events as well as expected conditions. In particular, the considerations must take into account the potential for igneous activity at the Yucca Mountain site; although such activity is improbable, it is not precluded by current information.[1] Therefore, the considerations of the ability of a repository system at Yucca Mountain to protect individuals must take into account the waste isolation characteristics in the event of such igneous activity.

Two types of pathways have been considered for igneous activity (DOE 2002, Section 3.1.2.4, p. 3-20). In the first, igneous activity occurs with low probability and magma intrudes into the emplacement drifts and disrupts the engineered barriers. In this case the waste isolation capabilities of these engineered barriers are diminished or eliminated altogether. In this igneous activity groundwater release scenario, the waste isolation characteristics of the natural barriers must be sufficient that the expected risk to the reasonably maximally exposed individual meets the performance objective. The second pathway is by way of eruptive release. In this case, the event occurs, again with low probability, and a fraction of the waste is erupted to the surface. The waste isolation characteristics of the repository system must be sufficient to limit the expected risk such that the individual protection limit can be met even for this event.

The particular waste isolation characteristics of the repository system differ in each scenario. The particular characteristics that apply to the nominal scenario (i.e., the scenario in which igneous activity does not occur), the igneous activity groundwater release scenario, and the igneous activity eruptive release scenario are delineated in the following sections.

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[1] The requirements of 10 CFR Part 63.113 also address the potential for inadvertent human intrusion. However, a scenario for such an event is not explicitly included here because it is unlikely that a driller would be able to drill through the waste package in the first 10,000 years and not be aware of the waste package. Nevertheless, the effect of inadvertent human intrusion on the conclusions of this evaluation is considered in Section 3.5.

2.1.1 Nominal Scenario

Under expected conditions, degradation of the waste packages would be very slow because of the corrosion-resistant material that would be used for the waste package outer barrier. General understanding of the corrosion resistance of the waste package materials, supported by measurements, indicates that the waste package would last much longer than 10,000 years under the full range of conditions expected deep below the surface at the Yucca Mountain site (BSC 2001a, Section 7.4.1, p. 7-72). Waste packages could fail earlier than expected due to fabrication defects. The rate of degradation of the waste package and the potential fabrication defects are expected to be important factors in the estimate of mean annual dose.

The drip shield over the waste package is also a factor in this estimate. The drip shield prevents seepage from contacting the waste even if the waste packages are breached. Consequently, advective release of radionuclides due to the seepage is limited while the drip shield remains intact. However, the drip shield does not prevent diffusive release of radionuclides from waste packages that may be breached. Diffusion of radionuclides from the breached waste packages, through the drift invert below the waste package, and into the rock is expected to contribute only a small amount to the mean annual dose. Therefore, the drip shield plays a role in meeting the individual protection requirements in the event waste packages fail before 10,000 years.

Radionuclide release from breached waste packages is limited by the rate at which the water contacting the waste can dissolve the waste form. In addition, the release is affected by the factors that determine the concentrations of radionuclides dissolved in the water or that are associated with colloids in the water. These factors combine to limit the amount of radionuclides that could migrate to the accessible environment to a small fraction of the potential inventory. The migration of this small fraction is affected by the characteristics of the natural barriers that retard the migration and disperse concentrations. Finally, the effect of these concentrations on an individual in the accessible environment is determined from the biosphere dose conversion factors.

The factors that affect the estimate of mean annual dose for the nominal scenario are therefore expected to include some or all of the following:

- The number of waste packages breached before 10,000 years
- The degree to which these waste packages are breached
- Performance of the drip shield in limiting advective release of radionuclides from breached waste packages
- Waste form dissolution
- Dissolved radionuclide concentrations and colloid-associated radionuclide concentrations
- Retardation and dispersion in the natural barriers
- Biosphere dose conversion factors associated with groundwater release.

2.1.2 Igneous Activity Groundwater Release Scenario

Normal degradation of the engineered barriers is accounted for in the nominal scenario. In the igneous activity groundwater release scenario, waste packages and drip shields are disrupted by intruding magma, exposing the waste to groundwater percolating down through the rock. Radionuclide release from the solid waste form is limited by the rate at which the water contacting the waste can dissolve the waste form. Again, the release cannot result in concentrations exceeding those defined by the controls on solubility of dissolved radionuclides and on colloid-associated concentrations. As in the case of the nominal scenario, the characteristics of the natural barriers that affect migration of the radionuclides also come into play.

The factors that affect the estimate of mean annual dose for the igneous activity groundwater release scenario are therefore expected to include some or all of the following:

- The probability of igneous intrusion into the emplacement drifts
- The number of waste packages and drip shields breached as a result of the intrusion
- The degree to which these waste packages are breached and permit release of radionuclides
- Waste form dissolution
- Dissolved radionuclide concentrations and colloid-associated radionuclide concentrations
- Retardation and dispersion in the natural barriers
- Biosphere dose conversion factors associated with groundwater release.

2.1.3 Igneous Activity Eruptive Release Scenario

The estimate of the mean annual dose for the igneous activity eruptive release scenario depends upon the amount of waste that is erupted with the tephra into the atmosphere, atmospheric transport of this material downwind (including its dispersion during transport), the deposition of contaminated tephra, and uptake by humans of radionuclides in the deposited tephra. The magnitude of the mean annual dose is proportional to the number of waste packages intercepted by the erupting magma. Because the performance measure is expected risk, the estimate of mean annual dose is also proportional to the probability of the event. Finally, the estimate is proportional to the biosphere dose conversion factors that are used to translate the concentrations of radionuclides in the deposited tephra into mean annual dose.

The processes that transport the radionuclides from the point of the eruption to the point of deposition also affect the estimate of mean annual dose. These include the wind speed and direction. They also include the magnitude of the eruption (as manifested in the volume of erupted material) and the size and density of the erupted particles. While these factors do play a role, their effect is expected to be less important than the amount of waste exposed by the eruption, the probability of this exposure, and the biosphere dose conversion factors.

2.2 CHARACTERISTICS AFFECTING GROUNDWATER PROTECTION

Waste isolation also includes inhibiting the migration of radionuclides to the accessible environment such that groundwater concentrations would not exceed the requirements of 10 CFR 63.113(c). The considerations in this regard are limited to likely features, events, and processes; e.g., they do not take into account unlikely igneous activity.

The Yucca Mountain site suitability evaluation considered the potential groundwater concentrations of radionuclides that might be released from the repository. These estimates indicate that concentrations of radium-226 and radium-228 that might be released from the repository would be orders of magnitude below the natural levels of these radionuclides, which are, in turn, well below the regulatory limit (BSC 2001b, Figure 4.1-16, pp. 4F-22 to 23). The estimate of gross alpha activity from released radionuclides is also orders of magnitude below natural levels of these radionuclides, which are also well below the regulatory limit (BSC 2001b, Figure 4.1-17, pp. 4F-24 to 25). The calculated dose from beta- and gamma-emitting radionuclides is orders of magnitude below the regulatory limit that applies in this case. The factors affecting these estimates are the same as those affecting the estimate of mean annual dose for the nominal scenario (see Section 2.1.1).

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3. SENSITIVITY STUDIES

The objective of the studies presented in this section is to provide a quantitative evaluation of the roles of the various components of the TSPA model in the assessment of waste isolation that would be provided by a repository system at Yucca Mountain. Sensitivity studies conducted for the site recommendation (CRWMS M&O 2000a and BSC 2001b) can be used to support such an evaluation. Those analyses were conducted using the TSPA-SR model (CRWMS M&O 2000b). However, the TSPA model has evolved somewhat since those sensitivity analyses were conducted. Changes include adoption of the definition of the boundary of the accessible environment and the concept of a reasonably maximally exposed individual to be consistent with the definitions of 10 CFR Part 63 and the addition of early waste package failures in the nominal scenario. Because these changes could modify the role of particular components under some circumstances, additional sensitivity studies are reported here in order to provide an adequate base of information for evaluating these roles.

The specific role of TSPA model components is explored through studies of effects on the estimate of mean annual dose in the first 10,000 years after permanent closure. Although the postclosure performance objectives address groundwater concentrations as well as annual dose, the information in Section 2.2 indicates that both quantities are measures of groundwater release of radionuclides from the repository and, as such, reflect the same general sensitivity to the roles of the various TSPA model components in limiting this release for the groundwater release scenarios.

As indicated in Section 1.2, the estimate of mean annual dose is the measure of expected risk considered in the regulations. Therefore, the base case for these sensitivity studies does represent an estimate of expected risk. However, the mean annual dose estimates for variations of the TSPA model components do not represent variations in expected risk because the probability of the variations is not taken into account. Therefore, although the curves calculated in each case are probability-weighted estimates, they do not include the full probability considerations. Accordingly, none of them should be used to indicate whether the individual protection limit of 10 CFR Part 63 might or might not be met under such variations. These studies are conducted only to provide insights into the potential significance of particular TSPA model components to expected risk.

3.1 TSPA MODEL FOR THE SENSITIVITY STUDIES

The TSPA model used for these sensitivity studies (referred to as the "base-case" model in this report) is similar to the revised supplemental model used for the site suitability evaluation (Williams 2001). All sensitivity studies utilizing the base-case model are fully probabilistic, employing a Monte Carlo sampling approach. Results obtained using this base-case model are shown in Figure 1. The results in this case are the estimates of mean annual dose for the igneous activity eruptive release scenario, the igneous activity groundwater release scenario, and the nominal scenario.

Components of the TSPA model whose role in these estimates is studied in this report are summarized in Table 1. Validation of all of these components according to the applicable procedures has not yet been completed. However, these components are considered to be sufficiently mature that they can be used in assessing the roles of each of them in quantitative estimates of postclosure performance. Accordingly, this TSPA model is used in these studies to support the determination of the level of model validation that is needed.

The base-case TSPA model for these studies differs from the revised supplemental model in only three ways. These changes have been made to improve calculational efficiency for the large number of cases evaluated. The first difference is that the number of realizations used in the Monte Carlo sampling for the igneous activity eruptive release scenario is reduced; that is, the base-case model for this scenario is the same as the revised supplemental model except that a smaller number of realizations is calculated. The revised supplemental analyses calculated 5,000 realizations for the igneous activity scenarios while the studies reported in this document calculate only 300 realizations. Figure 2 compares the results using 5,000 realizations with those using only 300 realizations. The difference between these two cases is not significant. Consequently, utilization of only 300 realizations for this scenario appears to be adequate for assessing the relative role of the TSPA model components in this case.

The second difference between the base-case model for these sensitivity studies and the revised supplemental model is in the approach to the igneous activity groundwater release scenario. The approach to specifying the time of occurrence of the igneous intrusion is to sample over the calculational period (100,000 years). The approach for the revised supplemental analyses is to sample uniformly over this period. The total number of realizations is therefore divided up among the time steps, and only about 10 percent of the realizations are associated with the events occurring in the first 10,000 years. Of these, only the events at the end of the 10,000 years receive the full benefit of that fraction of realizations, and events occurring earlier benefit from a proportionally fewer realizations. In order to provide accurate representation of events occurring in the early period, a large number of realizations is necessary to ensure that the number of realizations in which the event occurs in the early period is large enough.

One way to deal with this problem is to increase the number of realizations occurring in the early period without increasing the total number, e.g., by weighting the sampling toward the early period. Figure 3 compares the result of the revised supplemental model linear sampling scheme (circles) with the result of a scheme in which the sampling is logarithmic with time (triangles). This approach to the sampling increases the estimate of mean annual dose in the early period because more contributions from the tails of the probability distributions (which could result in increased release of radionuclides) are captured. These additional contributions increase the estimate of mean annual dose over that from the linear sampling scheme.

The weighted sampling scheme approach still requires a large number of realizations to ensure a stable mean in the early period. Another way to deal with this problem is to assume that the variation in processes affecting release does not depend strongly on the time the event occurs. In this case, the same release curve can be used to represent system performance whenever the igneous intrusion event occurs. All realizations are then devoted to this single curve. The effect is to increase the ability to capture the full range of each probability distribution with a smaller number of realizations. Figure 3 shows the result of this approach using only 300 realizations (solid curve). The result is comparable to that for the logarithmic sampling approach. Some effects are not accurately accounted for in this approach (e.g., glacial maximum climate that occurs at a fixed absolute time and not simply at some time after the event occurs). However, the glacial maximum climate occurs at 39,000 years, so that the approach appears to be reasonable for estimating the effects in the first 10,000 years.

The third difference between the base-case model for these sensitivity studies and the revised supplemental model is in the treatment of early waste package failure due to improper heat treatment. Such early failure has very low probability; consequently, only a fraction of the realizations in a TSPA study will contain such a failure. The effect is that out of 300 realizations, fewer than 70 will involve any early waste package failure. This number of realizations is not sufficient to provide a reliable estimate of effects that come into play only when the early failure occurs. The approach used here is to force early failure of one package in every realization, increasing the number of realizations for the models and parameters that come into play when a waste package is breached. This approach results in a higher mean annual dose associated with these early failures. Figure 4 compares the results from this base-case model with those from the revised supplementary model. The nominal scenario result for the revised supplementary model is a factor of 0.26 smaller than the nominal scenario result for the base-case model. This difference is not likely to be significant in assessing the role of various TSPA model components in the estimate of mean annual dose. Consequently, the advantage of providing a more stable estimate of the mean annual dose without increasing the number of realizations is considered to outweigh the disadvantage of the overestimate of the mean annual dose in the early period.

These three changes have been adopted to expedite the sensitivity studies. They are not likely to affect conclusions regarding the relative importance of the different model components. However, those conclusions can be checked as needed later as the TSPA model is further developed.

The results in Figure 1 show that the total mean annual dose estimate is dominated by the igneous activity eruptive release scenario results. The peak mean annual dose estimate in this case is on the order of 0.1 mrem, more than two orders of magnitude below the individual protection limit of 15 mrem. The 10,000-year mean annual dose estimate for the igneous activity groundwater release scenario is less than 0.01 mrem, more than an order of magnitude below the eruptive release estimate and more than three orders of magnitude below the individual protection limit. The 10,000 year mean annual dose estimate for the nominal scenario is on the order of 0.0001 mrem or less, more than three orders of magnitude below the eruptive release estimate and more than five orders of magnitude below the individual protection limit. Figure 5 shows the dominant radionuclide contributors to the mean annual dose estimates for the igneous activity eruptive release scenario, the igneous activity groundwater release scenario, and for the nominal scenario.

3.2 QUALITY ASSURANCE

The activity evaluation conducted to determine the level of quality assurance required for this evaluation concluded that these sensitivity studies serve only to support management decision-making and are, therefore, not required to be subject to quality assurance requirements (BSC 2002). However, to ensure complete traceability of the studies, they were carefully controlled and documented. Each of the studies was prepared by a qualified originator and documented in sufficient detail that it can be verified without recourse to the originator. In addition, each study was conducted using a controlled master TSPA model, and changes from the master TSPA model were implemented in a controlled manner and fully documented.

A qualified checker different from the originator reviewed each study. All studies were checked for correctness and accuracy and the results of the check were documented. Following this review, further changes were locked out and the studies were approved and submitted to the Technical Data Management System. Documentation in this submittal includes the TSPA model file, the input files for the associated dynamic-link libraries, descriptions of each study, a checklist documenting the changes from the master TSPA model, and a flow chart showing the relationship between the sensitivity studies and base cases for those studies.

The sensitivity studies were performed using GoldSim Version 7.17.200 (BSC 2001c). This numerical code is the same as that used for the revised supplemental analyses for the site suitability evaluation (Williams 2001), the final environmental impact statement supplemental analysis (BSC 2001d), and the *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The documentation for all of the records, including the detailed description of the model variations, records of checking, and input and output files are stored in the Technical Data Management System with the Data Tracking Number MO0205MWDTSP09.002.

3.3 ONE-OFF SENSITIVITY STUDIES

The sensitivity studies discussed in the following sections assess the role of the TSPA model components in the estimate of 10,000-year mean annual dose.[2] The general approach considered in the section is "one-off" TSPA sensitivity studies. That is, studies are conducted in which parameters of a single TSPA model component are varied over a sufficient range to explore the role of that component in the estimate of mean annual dose, maintaining the representations of the other model components as in the base-case model. Table 1 indicates the particular sensitivity studies that apply to each of TSPA model components.

The importance of a particular TSPA model component to expected risk is assessed in terms of the potential for that component to affect the mean annual dose curve. As discussed in the introduction to Section 3, the change to the mean annual dose estimate resulting from a change in the TSPA model component does not necessarily correspond to a change in expected risk because the probability of the change has not been taken into account. Nevertheless, if the change in the mean annual dose estimate is insignificant, then it can be concluded that the effect of the TSPA model component being studied on expected risk is also insignificant. If the change in the mean annual dose estimate is significant, it is not necessarily true that the effect of the TSPA model component is significant. For this reason, if the change in the mean annual dose estimate is significant, the role of the TSPA model component being studied is described in this report as "potentially significant."

For the purpose of these studies, the potential significance of a TSPA model component is assessed in terms of whether changes in the component result in a change in the estimate of mean annual dose in the first 10,000 years comparable to the regulatory standard of 15 mrem. In the

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[2] The focus of the analyses is the 10,000-year performance objectives of 10 CFR 63.113; however, results are shown for 100,000 years to provide insights into trends that are initiated in the first 10,000 years but are not fully manifested until after this period.

absence of synergistic effects, a change in the estimate in a one-off study of less than 1 mrem would be considered to be insignificant in comparison with this limit. In those cases where a change in one TSPA model component could combine with changes in other components not evaluated in a given one-off study, a change in the estimate of less than 0.1 mrem is considered to be insignificant. Explicit consideration of combined effects of changes in several components at once is provided in Section 3.4.

3.3.1 Climate and Net Infiltration Sensitivity Study

The first study examines the role of the climate and net infiltration component of the TSPA model. The TSPA model includes this component to help determine the amount of water that could contact waste, mobilize radionuclides, and carry those radionuclides away from the repository to the water table.

Figure 6 examines the sensitivity of the estimate of mean annual dose to the climate and net infiltration model component. This figure compares the results of the base-case model with a model that is unrealistic but which provides extreme values to allow exploration of the role of the model. The extreme model provides an unsaturated zone flow field that is consistent with an infiltration flux of the same order of magnitude as the precipitation flux. Precipitation onto Yucca Mountain averages about 190 mm/year under current conditions and is expected to average more than 300 mm/year over the next 10,000 years (BSC 2001a, Table 3.3.1-1, p. 3T-1). The corresponding percolation flux in the base-case infiltration model averages about 4.6 mm/year under present day conditions and about 12 mm/year over the next 10,000 years (BSC 2001a, Table 3.3.2-1, p. 3T-5). The extreme model assumes a flow field associated with the highest infiltration rate for the glacial maximum climate. The infiltration flux in this case averages about 150 mm/year (BSC 2001a, Table 3.3.2-3, p. 3T-7), approximately an order of magnitude greater than the infiltration flux for the base-case model and of the same order of magnitude as the present-day precipitation on Yucca Mountain. This extreme infiltration is considered to ensure that the role of the infiltration model is adequately evaluated.[3]

The results for the nominal scenario in Figure 6 show little change to the estimate of mean annual dose. The drip shields remain intact for more than 60,000 years; therefore, the increase in infiltration does not translate into an increase in the amount of water contacting waste in the first 10,000 years. The effect of increased infiltration in this case is increased wetting of the drift invert and associated changes to its transport properties and in the flow below the repository that can transport radionuclides to the water table. The results for igneous activity groundwater release scenario show somewhat greater increase because drip shields are disrupted, permitting advective flow to contact the waste. The estimate of mean annual dose is dominated by the solubility-limited plutonium-239 and plutonium-240 (see Figure 5b) so that this increase does result in an increase in the mean annual dose estimate. However, even in this case the increase is less than 0.01 mrem and is not considered to be significant.

[3] Infiltration models intermediate between the base-case model and the extreme model considered here are expected to provide results between those shown here. That is, the increased flux of the analysis conducted here is so high that it addresses considerations of flow focusing or episodicity effects on the flow system. The effect on seepage of intermediate values for these factors is considered in Section 3.3.2.

These results indicate that the details of the climate and net infiltration models do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR model also show no significant impact of magnitude of the net infiltration or the details of the unsaturated zone flow field on the estimate of mean annual dose (CRWMS M&O 2000a, Section 5.2.1.1, P. 5-9; BSC 2001b Section 3.2.1, p. 3-3).

3.3.2 Seepage Sensitivity Study

The seepage component of the TSPA model represents the flow of water into the emplacement drift that is a primary determinant of the moisture conditions within the emplacement drift. The seepage flux determines the advective flow contacting the drip shield and the flow through breaches in the drip shield in the TSPA model. This model component is therefore a factor in determining the amount of water contacting the waste packages, the amount of water entering breached waste packages, and the moisture conditions in the drift invert.

Seepage is not the only source of water affecting these elements. The moisture in the drift invert is evaluated in thermal-hydrologic (TH) analyses that take into account temperature and moisture content of the air, as well as the seepage. In addition, the TSPA model assumes a volume of water present within the waste package even when no seepage occurs to account for equilibrium between the moisture in the rock, in the air, and in the waste. These factors affect the sensitivity of the estimate of mean annual dose to the TSPA model component for seepage.

Figure 7 compares the base-case results with the results for different seepage models. In the base-case model, the seepage associated with a specified percolation flux varies over a range appropriate to that flux. In addition, the base-case model accounts for focusing of the flow due to heterogeneity in the rock and episodicity in the flow system. The first alternative model uses the 95th percentile of the base-case seepage distribution for the calculated percolation flux, the 95th percentile of the base-case flow-focusing factor, and the 95th percentile of the base-case episodicity factor. The comparison between the base-case model and this alternative model are shown (only for the igneous activity groundwater release scenario) in Figure 7. The results do not show a significant difference between these models. As in the case of the study of the effects of increased infiltration, the changes for the nominal scenario are negligible because the drip shield diverts water away from the waste and the only effect of the seepage is to change the moisture conditions in the drift invert. The changes are somewhat larger for the igneous activity groundwater release scenario because drip shields and waste packages are breached and the waste is directly exposed to the water. The increase in seepage results in an increase in the release of the solubility-limited radionuclides. However, even in this case, the increase is not significant.

One possibility for the small impact of the change in the model is that the variation considered is not sufficient to explore the full range of possibilities. There could be intermediate values for flow focusing or episodicity that could result in larger effects. This possibility is addressed by considering a more extreme case. The base-case model provides zero seepage over approximately half the waste packages and an average seepage flux that is less than 0.1 m^3/year over the other waste packages (BSC 2001a, Table 4.3.1-1, p. 4T-1). The extreme model considers the effect of seepage of 1 m^3/year over the location of every waste package. Thus, the extreme model provides more than a factor of 10 increase in the flux, effectively focusing flux onto every waste package; this model is therefore considered to be adequate to assess the role of the seepage component of the TSPA model.

The results for the nominal scenario show no significant difference in the first 10,000 years (while the drip shields remain intact). Potential effects in the event there were no drip shields can be inferred by examining the radionuclides that dominate this release. Figure 5 shows that the radionuclides that dominate the nominal scenario releases are soluble radionuclides such as carbon-14 and technetium-99 whose release is controlled by the waste form dissolution rate. Increased seepage would therefore not affect the release of these radionuclides significantly. As a result, it can be concluded that, even in the event the drip shields were not included in the engineered barrier system, an increase in seepage would not significantly change the mean annual dose estimate for the nominal scenario.

The results for the igneous activity groundwater release scenario are somewhat different from those for the nominal scenario because the release in this case is dominated by the solubility-limited radionuclides, plutonium-239 and plutonium-240. The results in this case show an increase of approximately a factor of 10 between the two curves, reflecting the increase in mass release of the plutonium isotopes associated with the increase in the amount of flow contacting the waste. However, even with this increase, the estimate of mean annual dose does not exceed 0.1 mrem.

These results indicate that the details of the seepage model do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR model also show no significant impact of the seepage on the estimate of mean annual dose over the range investigated (CRWMS M&O 2000a, Section 5.2.1.2, p. 5-10; Section 5.3.1.2, p. 5-34; BSC 2001b, Section 3.2.2, p. 3-7).

3.3.3 In-Drift Environments Sensitivity Studies

The in-drift temperature, moisture, and water chemistry components of the TSPA model specifically refer to the environments within the drift invert. In the general sense, the in-drift environments also include those conditions that affect the drip shield and the waste package. However, these factors are addressed in the drip shield and waste package degradation components of the TSPA model.

The effects of moisture conditions in the drift invert are not explicitly evaluated here. These effects are considered indirectly in Sections 3.3.2 and 3.3.9. The sensitivity studies discussed in those sections suggest there is not a strong dependence on the amount of water present in the emplacement drifts either for the nominal scenario in which the drip shield remains intact for 10,000 years or for the igneous activity groundwater release scenario in which drip shields are breached. The effects of moisture, as manifested in the relative humidity have been explicitly considered in the Supplemental Science and Performance Analysis (SSPA) analysis (BSC 2001b, Section 4.1, p. 4-1ff). The results do not show a significant effect on the waste package degradation in the first 10,000 years. The SSPA analyses explicitly considered the importance of the relative humidity threshold for general corrosion and found that, although there is an effect on the starting times for general corrosion, there is no significant effect on the failure of waste packages in the first 10,000 years for the cases evaluated.

Likewise, the effects of in-drift temperature are not explicitly evaluated here. The role of temperature has been considered in the SSPA (BSC 2001b, Section 4.1, p. 4-1ff) in which the higher temperature operating mode and lower temperature operating mode were compared. This SSPA comparison actually addresses a broader range of effects than temperature in the emplacement drifts because the effects of temperature are explicitly manifested in a number of the TSPA model components. They are directly accounted for in the TH effects in the unsaturated zone flow, the seepage, and the in-drift moisture model components. They are indirectly represented in terms of thermal-mechanical and thermal-hydrologic-chemical effects on near-field rock properties. They are also taken into account in temperature dependence of processes within the waste package. The SSPA comparison of the two different thermal operating modes showed very little impact of the temperature over the range spanned by these two operating modes. Calculated near-field temperatures differed by as much as 60°C and yet the differences between the estimates of mean annual dose for the nominal scenario in the first 10,000 years were less than 0.0004 mrem and the effect on total mean annual dose was negligible (BSC 2001b, Figure 4.1-1, p. 4F-1).

These results strongly suggest that the in-drift thermal effects accounted for in the base-case TSPA model components do not significantly affect the estimate of mean annual dose in the first 10,000 years. Part of this conclusion is due to the fact that most of the model components affected by temperature do not play a strong role in the estimate of mean annual dose. For example, Sections 3.3.1 and 3.3.2 show that details of the flow and seepage have little effect on the estimate of mean annual dose. Accordingly, thermal changes to those components would also be unlikely to affect this estimate significantly. Any temperature dependencies in those base-case model components that are important to postclosure performance are such that changes in drift wall temperatures from 130° to 70°C or changes in peak waste package temperatures from 160° to 80°C do not have a significant effect.

The remaining consideration is the effect on performance of water chemistry, in particular, the chemistry in the drift invert. The main effect of the water chemistry in this case is on the concentrations of radionuclides in the invert. In particular, the pH and ionic strength of the drift invert could affect the solubility limit of dissolved radionuclides and the colloid-associated radionuclide concentrations in the drift invert. These two aspects of the in-drift water chemistry are evaluated in Figures 8 and 9.

The results in Figure 8 address the sensitivity to pH of the water in the drift invert. The base-case TSPA model for invert pH ranges between 4.8 and 7.3 in the first 4,000 years and between 7 and 8 after 4,000 years (BSC 2001b, Figure 4.2.4-1, p. 4F-38). Figure 8 shows the estimate of the nominal scenario mean annual dose calculated using fixed pH values of 3, 7, and 10, as well as the base-case result. These results do not show a significant effect on the estimate of mean annual dose over this range. Although the pH affects the concentration limits for key radionuclides, the change is not sufficient to result in a significant change in the mean annual dose estimate of the range of pH evaluated.

The results in Figure 9 address the sensitivity of the mean annual dose estimate to ionic strength in the drift invert. The studies in this case consider ionic strengths well outside the range considered in the base-case model. In addition to the base-case results, Figure 9 shows the results for ionic strengths that are a factor of 1.5 greater than the upper bound of the range considered in the base case and a factor of 1.5 lower than the lower bound of that range. The main effect of the ionic strength in the TSPA model is on the colloid concentration limits, and therefore on the concentrations of radionuclides associated with these colloids. However, even with this increase, the colloid concentrations are insufficient to affect the estimate of mean annual dose that is dominated by the dissolved radionuclide concentrations.

The in-drift chemistry could also affect radionuclide concentrations indirectly, by affecting the in-package chemistry. The effect of in-package chemistry is examined directly in Section 3.3.5.

These results indicate that the details of the in-drift chemistry model do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR model also show no significant impact of the invert chemistry on the estimate of mean annual dose over the range investigated (BSC 2001b, Section 3.2.4.2, p. 3-16).

3.3.4 Engineered Barriers Performance Sensitivity Studies

The TSPA model components studied here are the models for performance of the drip shield and the waste package. These components both play a role in determining the exposure of the waste to water, and the waste package performance model plays a role in estimating the containment provided by the waste package and the release of radionuclides from the repository.

Figure 10 compares the base case with results of two studies in which the model component for drip shield performance is varied from the base-case representation. Only the nominal scenario is considered in this case because the drip shields are disrupted in the igneous activity groundwater release scenario and degradation processes play no role. The first study is for a case in which the general corrosion rate of the drip shield material is enhanced by a factor of more than five.[4] This enhanced rate is outside the range of uncertainty represented in the base-case model to address aleatoric uncertainty in the corrosion rate due to temperature and water chemistry variations, as well as epistemic uncertainty in the corrosion rate itself. However, even for this high rate, the results show that none of the drip shields are calculated to fail before 10,000 years and the estimate of mean annual dose in the first 10,000 years is not affected.

The second study presented in Figure 10 considers a more extreme case in which failure of the drip shield is forced to occur before 10,000 years. In this case the barrier function of the drip shield is completely "neutralized," i.e., all drip shields are assumed to be completely breached at the time of permanent closure. This is an unrealistic model that is considered in order to provide

[4] The median of the measurements of the drip shield general corrosion rate in weight-loss experiments is about 0.025 microns/year and the 95th percentile value is about 0.12 microns/year (CRWMS M&O 2000d, Section 6.5.4, p. 59). The measured corrosion rate is corrected for deposition of silicates during experiment by adding a factor that ranges between 0 and 0.063 microns/year. The approach for this analysis is to utilize the 95th percentile of the uncorrected measurements and to increase it by 0.17 microns/year to ensure the analysis goes beyond the range of the correction factor. The effective general corrosion rate is 0.29 microns/year, more than a factor of 10 greater than the median of the measured values and more than a factor of 5 greater than the median corrected value.

additional insight into the drip shield performance model. The result for this drip shield neutralization case in Figure 10 shows an increase in the estimated mean annual dose because the drip shield provides no barrier to water contacting the waste packages. This flow affects the rate of release of radionuclides from the waste packages that are breached. With the drip shield in place, no advective flow contacts the waste, and the release of radionuclides from the breached waste packages in the model is purely diffusive. Without the drip shield, advective release is also possible, and it is this additional advective component that results in the increase from the base-case result.

The constituents of this advective component are shown in Figure 11. This figure compares the contributions of mobile radionuclides (carbon-14 and technetium-99) and less mobile radionuclides (neptunium-237, plutonium-239, and plutonium-240) for the base case and for the case with the drip shield neutralized. The base case is dominated by diffusive release of the mobile radionuclides. The diffusion coefficient for the less mobile radionuclides is much smaller than that for the more mobile radionuclides and results in negligible diffusive release of these radionuclides in the first 10,000 years. When the drip shield is neutralized, advective release is permitted. The additional advective component for the mobile radionuclides is much larger than the diffusive component in the first 10,000 years. Likewise, the advective component for the less mobile radionuclides is not only larger than the diffusive component for those radionuclides, it is more than a factor of ten greater than the base-case contribution of the mobile radionuclides. However, the total advective release for the neutralized drip shield case is less than 0.1 mrem and is considered to be insignificant.

The results of the studies presented in Figure 10 indicate that the details of the drip shield degradation model do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses using the TSPA-SR model also show no significant impact of the drip shield degradation model on the estimate of mean annual dose over the range investigated (CRWMS M&O 2000a, Section 5.3.3.1, p. 5-35).

Figure 12 compares the base case with results of three studies in which the model component for waste package performance is varied from the base-case representation. The base-case degradation rate includes early failure of a small fraction of the waste packages due to improper heat treatment during manufacturing and later degradation of the waste packages due to corrosion. The base-case degradation model includes a probability distribution for general corrosion and a stress corrosion cracking (SCC) model whose parameters are also distributed to reflect the uncertainty and variability in the current information. The results using the base-case model show two effects. First, they show the release associated with early waste package failure (modeled as discussed in Section 3.1). They also show an increase in the mean annual dose after 60,000 years associated with further degradation of the waste package. The three studies in Figure 12 examine the role of this degradation model.

In the first study, the general corrosion rate is changed to a single value that is the 95th percentile value of the base-case general corrosion rate probability distribution. The base-case median corrosion rate is less than 0.05 microns per year and the 95th percentile general corrosion rate is 0.08 microns per year. The degradation model for this study also includes an extreme model for SCC degradation: the residual stresses in the weld region of the outer and middle lids weld region are both set to their 95th percentile values, and the number of defects in the closure weld is set to its 95th percentile value. The results using the fixed, enhanced degradation rate are shown by the circles in Figure 12. This degradation model still results in no failure of waste packages before 10,000 years beyond the base-case early failures due to improper heat treatment and, consequently, no change in the estimate of mean annual dose in the first 10,000 years.

In the second study, the general corrosion rate is increased well outside the range supported by the current information. In this case, the general corrosion rate is multiplied by a factor of eight[5], providing rates that are higher than any that have been observed. The effect on the estimate of mean annual dose is shown by the triangles in Figure 12. In this case, waste package failures due to corrosion begin before 10,000 years, and the estimate of mean annual dose increases correspondingly. However, even for this extreme degradation model, the estimate of mean annual dose in the first 10,000 years does not exceed 0.0001 mrem and is calculated to exceed the individual protection standard (15 mrem) only after 40,000 years.

The third study in Figure 12 examines the effects of neutralizing the waste package entirely. In this case all waste packages are assumed to be 100 percent breached at the beginning of the simulation. As in the case of the drip shield neutralization shown in Figure 10, this approach is not a realistic case but is conducted to increase insight. The results are shown as the upper curve (squares) in Figure 12. Figure 13 shows the contributions of the mobile and less mobile radionuclides for this case. The release from the engineered barrier system in the first 10,000 years is purely diffusive since the drip shields are still intact and prevent advective flow through the system. The estimated mean annual dose associated with this diffusive release is more than 10 mrem. This change is significant and indicates the potential significance of the waste package model to expected risk.

The results in Figure 12 are consistent with the results using the TSPA-SR model. Analyses using the TSPA-SR model suggest that the waste package plays an important role in the estimate of mean annual dose over the range investigated (CRWMS M&O 2000a, Section 5.2.3, p. 5-12 and Section 5.3.3.2, p. 5-36; and BSC 2001b, Section 3.2.5.1, p. 3-18; Section 3.2.5.2, p. 3-19; Section 3.2.5.3, p. 3-20; and Section 3.2.5.4, p. 3-21).

3.3.5 In-Package Environments Sensitivity Studies

In-package environments include the temperature, moisture, and water chemistry within the waste package to which the waste form is subjected. The effects of moisture conditions in the waste package are not explicitly evaluated here. In addition to advective flow through breached waste packages, the base-case model includes an in-package volume of water associated with hygroscopic effects and wetting of waste form degradation products by the advective flow. The

[5] The base-case general corrosion model is derived from the results of 24-month weight-loss measurements. The mean general corrosion rate from these measurements is about 0.01 microns/year, and the 95th percentile value is about 0.04 microns/year (CRWMS M&O 2000e, Section 6.91, p. 105). For the waste package degradation model, the measured corrosion rate is corrected for deposition of silicates during the experiment by adding a factor that ranges between 0 and 0.06 microns/year. The factor of eight results by considering the 95th percentile of the combination of 6-month and 12-month measurements (0.11 microns/year), the maximum silicate deposition correction of 0.06 microns/year, and the maximum enhancement factors for thermal aging and microbial effects. The net effect of this approach is a factor of eight multiplier on the corrected and enhanced corrosion rate of the base-case model.

base-case model assumes uniform mixing of mobilized radionuclides in this volume. This representation is conservative and tends to limit sensitivity to changes in the moisture conditions in the TSPA model.

Sensitivity to in-package temperature in the TSPA model is explicitly evaluated here. The base-case model does not account for any temperature difference between the waste form and the waste package wall. The effect of this approach is evaluated in Figure 14. This figure compares the results of a study in which a difference between the waste form temperature and the waste package wall temperature is explicitly taken into account. In this case the waste form temperature is 60°C greater than the waste package wall temperature at the time of emplacement. The increment decreases to zero at 100,000 years (the decrease results in an increment of 54°C at 10,000 years). Although the cladding creep degradation model and the waste form dissolution model both depend on temperature, the increment considered here does not result in any significant change to the estimate of mean annual dose because the change in waste form degradation associated with this increased temperature is not significant. Section 3.3.6 provides additional insight into the sensitivity of the estimate of mean annual dose to waste form degradation.

A second study examines the role of the in-package chemistry component of the TSPA model. This component is included in the TSPA model to ensure that chemical effects on waste form degradation and radionuclide concentrations are taken into account.

In the first 10,000 years, the base-case pH values for the nominal scenario range between 3.4 and 7.7 in the base-case model (BSC 2001a, Tables 9-3, 9-4, and 9-5, pp. 9T-1 and 9T-2). Figure 15 shows the results for fixed pH values of 3, 7, and 10, as well as the base-case results. The results for the nominal scenario are changed very little over this range. Figure 5(c) shows that the radionuclides that dominate the estimate of mean annual dose for the nominal scenario are those whose concentrations are controlled by the waste form degradation rate. These small increases reflect the limited dependence of the waste form degradation rate on the water chemistry in the TSPA model.

The base-case pH values for the igneous activity groundwater release scenario have a different range than those for the nominal scenario because the model assumes that the relevant water chemistry corresponds to that at the drift wall (drip shields and waste packages are disrupted by the intruding magma). The base-case pH in this case ranges between 7 and 9 (CRWMS M&O 2000f, Section 3.3.5). The variation over pH values from 3 to 10 therefore has a larger effect for this scenario than for the nominal scenario. The main effect in this case is on the solubility of plutonium-239 and plutonium-240, the solubility-limited radionuclides that dominate the estimate of mean annual dose for this scenario. Figure 16 shows the dependence of the neptunium and plutonium solubility limits on the pH of the water in the waste package. This figure shows that the solubility can vary by orders of magnitude over this range and this change produces the change in the estimate of mean annual dose for this scenario. Figure 17 illustrates the effect of this range in pH by showing the changes in the contributions of neptunium-237 and plutonium-239 over this range.

Figure 18 shows the role of the ionic strength of the water in the waste package. The main effect of the ionic strength taken into account in the TSPA model is in defining the colloid concentrations and, therefore, the concentrations of radionuclides sorbed to these colloids. The study considers results for ionic strengths that are a factor of 1.5 greater than the upper bound of the range considered in the base case and a factor of 1.5 lower than the lower bound of that range. As in the case of the considerations of in-drift chemistry, the model for ionic strength has no significant impact on the estimate of mean annual dose.

Although these results do show changes, the changes are less than 0.1 mrem. These results therefore indicate that the details of the in-package chemistry model do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR model also show no significant impact of the in-package chemistry on the estimate of mean annual dose over the range investigated (BSC 2001b, Section 3.2.7.1, p. 3-25).

3.3.6 Waste Form Performance Sensitivity Studies

The waste form performance component of the TSPA model is used to describe the degradation of the commercial spent nuclear fuel (CSNF) cladding and dissolution of the waste form matrix. Both of these effects are considered here.

Figure 19 shows the results of two studies of the role of the CSNF cladding in the estimate of mean annual dose. In the first study the parameters of the cladding degradation model are set to their 95th percentile values to examine the sensitivity to this degradation rate. That is, the fraction of fuel rods with initially failed cladding is set to its 95th percentile value; the unzipping rate is calculated by setting the fuel alteration rate and the conversion factor to their 95th percentile values; and the localized corrosion rate and SCC failures are set to their 95th percentile values. The resulting calculation shows a very small change to the base-case result.

The second study is more extreme. In this case, 100 percent of the cladding is assumed to be initially breached. The results for this case are also shown in Figure 19. These results show an increase in the nominal scenario release of about one order of magnitude due to the increased exposure of the CSNF waste to water. The resulting change to the estimate of mean annual dose is less than 0.001 mrem and is therefore not significant.

These results indicate that the details of the cladding degradation model do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR model also show no significant impact of the cladding on the estimate of mean annual dose over the range investigated (CRWMS M&O 2000a, Section 5.3.4.1, p. 5-38; BSC 2001b, Section 3.2.7.2, p. 3-26).

A third study is conducted to evaluate the role of the waste form dissolution rate, apart from the role of the cladding. The results of the study of the waste form dissolution rate are shown in Figure 20. Three types of waste forms are considered in the TSPA model, namely, DOE spent nuclear fuel, high-level waste (HLW) glass, and CSNF. Since complete degradation of DOE spent nuclear fuel occurs in one time-step in the model, no sensitivity study is performed on this parameter. For HLW, each parameter of the glass degradation rate model is set to its 95th or 5th percentile value to maximize the rate. The dissolution rate of the CSNF is set to its 95th

percentile value. The results in Figure 20 show only a small effect on the estimate of mean annual dose for the groundwater release scenarios and no significant impact on the estimate of total mean annual dose. The dissolution rates of HLW and CSNF are so high in the base case that increasing them does not have a significant impact on the estimate of mean annual dose.

3.3.7 Dissolved Radionuclide Concentrations Sensitivity Studies

The dissolved radionuclide concentrations component of the TSPA model determines the solubility limits for radionuclides as a function of temperature and chemistry of the water in the package and in the drift invert. Because of the dependence of these limits on chemistry, the role of this component has already been explored to some extent in the studies of the in-drift and in-package chemistry components (see Sections 3.3.3 and 3.3.5). However, those studies do not directly examine the role of the solubility limits. This study is therefore conducted to examine this sensitivity directly.

Figure 21 shows the results of specifying the solubility limits for protactinium, neptunium, plutonium, technetium, and thorium at their 95th percentile values both in the waste package and in the drift invert. The results indicate little sensitivity to the solubility over the range considered in the current model. This limited sensitivity is explained in part by the fact that protactinium and thorium do not contribute significantly, and technetium is so soluble that its release rate is limited by the waste form dissolution rate. However, even the variation for plutonium and neptunium considered here does not affect the estimate of mean annual dose significantly.

Studies are conducted to explore a wider range of solubility for neptunium and plutonium. Figure 22 shows the results for a case in which the neptunium solubility is varied from 10^5 mg/L to 10^{-3} mg/L, and the plutonium solubility is varied from 10^5 mg/L to 10^{-7} mg/L. These ranges span the full range of possible solubilities indicated in Figure 16.

The results for the nominal scenario in Figure 22 show little sensitivity to the solubilities of these two radionuclides. Technetium-99, iodine-129, and carbon-14 in this case determine the mean annual dose estimate, and the variation of neptunium and plutonium solubilities has little effect. The results for the igneous activity groundwater release scenario, however, are dominated by plutonium-239 and plutonium-240. As a result, the variation in the solubility of these isotopes has a large relative effect on the estimate of mean annual dose for that scenario. The change is on the order of 0.1 mrem. A larger change could occur for conditions under which the plutonium solubility is increased above the range considered here. Accordingly, it is possible that this factor could be important to the estimate of mean annual dose.

These results indicate that the details of the dissolved radionuclide concentration model for plutonium could be important to the estimate of mean annual dose under some conditions. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR also show moderately important impacts of the plutonium solubility limit on the estimate of mean annual dose over the range investigated (CRWMS M&O 2000a, Section 5.2.4.2, p. 5-18; Section 5.3.4.2, p. 5-39; BSC 2001b, Section 3.2.7.3, p. 3-27).

3.3.8 Colloid-Associated Radionuclide Concentrations Sensitivity Studies

These studies examine the role of the colloid-associated radionuclide concentration component of the TSPA model. Movement of colloids through the system represents a way in which some radionuclides may be transported more rapidly to the accessible environment than as dissolved species, and this component is included in the TSPA model to ensure that the contribution of such radionuclide transport is taken into account. The studies explore the role of this model component in defining the concentrations of the colloid-associated radionuclides.

The TSPA model addresses three different types of colloids: clay and silica colloids occurring naturally in the groundwater, iron-(hydr)oxide colloids associated with degradation of waste package materials, and colloids derived from degradation of the HLW glass waste form. The model accounts for attachment of radionuclides onto these colloids by sorption and embedding of radionuclides in the HLW inventory in the waste form colloids. The largest contributor to the estimate of mean annual dose from these radionuclides is from the plutonium isotopes associated with the colloids generated during degradation of the HLW glass. Figure 23 shows the effect of artificially increasing the concentration of waste form colloids by a factor of 100 (thus, the mass of plutonium increases even though the concentration of plutonium on the colloids does not change). This increase could affect the concentrations of radionuclides that could sorb to the waste form colloids; however, the results in Figure 23 show a negligible effect on the estimate of mean annual dose for the nominal scenario. The change is also negligible for the igneous activity groundwater release scenario in the first 10,000 years. There is change after 10,000 years associated with a change in the water chemistry model at 10,000 years in the base-case in-package chemistry model. Even in this case, however, the estimated mean annual dose only reaches about 0.2 mrem at about 17,000 years.

Figure 24 shows the result of increasing the concentrations of plutonium and americium modeled as irreversibly sorbed to the waste form colloids. In this study these concentrations are set to values that are a factor of 1.5 times the maximum value possible in the base-case model. In addition, the colloids are assumed to be stable, i.e., the effects of ionic strength and pH on colloid stability are ignored. The effect of these changes is to increase the concentration of plutonium and americium above any value calculated in the base-case model. The resulting change in the estimate of mean annual dose is not significant.

Finally, Figure 25 shows the sensitivity to a different part of the colloid concentration model. The base-case model assumes limits to the concentration of colloids that are affected by pH and ionic strength of the water. These limits are based on observations, e.g., the increase in colloid instability with increasing ionic strength. The results in these figures show the sensitivity of the results to these limits. In this case, the concentrations of the colloids are artificially increased to values that are a factor of 100 greater than the upper limit of the base-case model. In addition, the concentrations of plutonium and americium on these colloids are also increased by a factor of 100. The result is more than a 10,000-fold increase in the colloid-associated concentrations of these radionuclides in the water. The estimate for the igneous activity groundwater scenario in Figure 25 is changed substantially. The change in the concentrations is so large that it swamps the effects of the chemistry that caused the peak between 10,000 years and 17,000 years in the igneous activity groundwater release scenario in Figure 23. Even so, the net effect is an increase in the mean annual dose estimate that is less than 0.1 mrem. This increase is due entirely to the increase in colloid-associated plutonium-239 and plutonium-240 release. The range considered here is so large that it is unlikely that conditions could arise that lead to even larger effects. Accordingly, this model is not considered to play a significant role in the estimate of mean annual dose.

The results of this section indicate that the details of the model for colloid-associated concentrations do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR also show no significant impact of the colloid concentrations on the estimate of mean annual dose over the range investigated (BSC 2001b, Section 3.2.7.4, p. 3-20).

This study only considers the sensitivity to colloid-associated radionuclide concentrations. The sensitivity to transport characteristics of colloid-associated radionuclides is examined in the study of the natural barriers radionuclide transport model in Section 3.3.10.

3.3.9 Engineered Barriers Radionuclide Transport Sensitivity Studies

These studies evaluate the role of the component for radionuclide transport through the engineered barrier system in the TSPA model. The current TSPA model considers both advective and diffusive transport of radionuclides out of the waste package, through the drift invert, and to the unsaturated zone rock. Advective transport only occurs when the drip shield is breached and permits water to flow onto and into the waste package. Diffusive transport occurs whenever the waste package is breached, both before and after the drip shield is breached. The comparison in Figure 10 between the results for the case where the drip shield remains intact (only diffusive release occurs) and the results for the case where the drip shield is neutralized (advective release also occurs) suggests the advective component is much more important than the diffusive component after the drip shields are breached.

Figure 26 examines the sensitivity of system performance to the parameters of the model for advective transport through the engineered barriers. The base-case model uses a range of sorption coefficients for transport out of the waste package and in the drift invert. The result of using the 5th percentile from this range is compared with the base-case result in this figure.[6] The effect of sorption is to delay the migration of the radionuclides that sorb onto the drift invert material. The result of using the smaller sorption coefficient is to reduce travel time and dispersion of the radionuclide concentrations during migration. The effect of this change is very small in Figure 26. Consequently, details of the model affecting retardation of radionuclide transport out of the waste package and through the drift invert are not significant to the estimate of mean annual dose.

Figure 27 shows the sensitivity of the mean annual dose estimate to the diffusivity of the drift invert. In this case the diffusivity is set to the largest value considered in the TSPA-SR analyses, i.e., the exponent of the saturation dependence is set to the 95th percentile of the distribution that has been measured for tuff gravel. The sensitivity of the estimate of mean annual dose is very weak for this range of values.

[6] Sorption (e.g., onto waste package internals and corrosion products) that could reduce aqueous concentrations of radionuclides within the waste package is not taken into account in the base-case model for these analyses. Its potential role in the TSPA model is evaluated in the SSPA analyses (BSC 2001b, Figure 3.2.8-2, p. 3F-39).

These results indicate that the foregoing details of the invert diffusion model do not play a significant role in the estimate of mean annual dose. This result is consistent with the results using the TSPA-SR model. Analyses of the nominal scenario show that, over the range of diffusivity in the TSPA-SR model, the results provide no significant impact of the diffusivity on the estimate of mean annual dose over the range investigated (CRWMS M&O 2000a, Section 5.2.5.1, p. 5-19; Section 5.3.5, p. 5-40).

The fact that the details of the advective and diffusive components of the transport model are not important to the estimate of mean annual dose implies that the model for determining the ratio of these two components is also not important with respect to transport in the engineered barrier system. However, this ratio could be important to transport in the unsaturated zone. In the base-case model, the slow diffusive release from the engineered barrier system is transferred into the matrix of the near-field rock while the advective release is transferred into the near-field rock fracture system. This ratio could therefore play a role if the transport characteristics of the rock matrix and fracture system of the unsaturated zone are very different. The effect of this ratio in this regard is considered in the next section.

3.3.10 Natural Barriers Radionuclide Transport Sensitivity Studies

These studies examine the role of components for transport of radionuclides through the unsaturated zone and saturated zone after release from breached waste packages. These components are examined together because the unsaturated zone and saturated zone barriers are complementary. That is, understanding of the role of each of these barriers in the TSPA model requires understanding of the role of the other. Consequently, these two barriers are considered together.

Up to 26 different radionuclides are considered in the TSPA (CRWMS M&O 2000g, Table 9). Screening arguments indicate that other radionuclides would not provide a significant contribution to annual dose or groundwater concentrations. However, while the model components for radionuclide transport in the engineered barrier system and in the saturated zone include all of these radionuclides, the model component for radionuclide transport in the unsaturated zone does not explicitly take into account two of them, strontium-90 and cesium-137. Computational demands imposed by other model components limit the number of radionuclides that can be considered in this component. Because the half-lives of these two radionuclides are relatively short (on the order of 30 years), they are likely to decay away before they reach the water table and are therefore unlikely to contribute significantly to mean annual dose. Consequently, the computational constraints are met by omitting strontium-90 and cesium-137 in favor of longer-lived radionuclides. The effect of this step is shown in Figure 28. In this study, two radionuclides with negligible effect on estimated mean annual dose in the base-case estimates are replaced by strontium-90 and cesium-137. The resulting estimate of mean annual dose differs somewhat from the base case when the failure associated with improper heat treatment is beginning. In the very early period only a few of the 300 realizations include a waste package failure and the number is too small for a stable mean. Consequently, the results differ somewhat in this early period. After about 3,000 years, stability of the mean sets in because all 300 realizations include a waste package failure.

The combined role of the unsaturated zone and saturated zone barriers is evaluated in Figures 29 and 30. In Figure 29, the combined effect of the nominal and igneous activity groundwater release is shown for a case in which both of these barriers are "neutralized." That is, the release from the engineered barrier system is discharged directly into the water usage volume at the accessible environment, and the unsaturated zone and saturated zone each provide no barrier to radionuclide transport. As in the case of the neutralization analyses for the engineered barriers (see Section 3.3.4), these neutralizations do not represent realizable situations but are evaluated only to provide insights into the roles of the barriers.[7]

Comparison with the base-case results for the combined groundwater release scenarios shows a large difference from the neutralization results, indicating that these barriers do affect the estimate of mean annual dose for the groundwater release scenarios. Neutralizing these barriers results in an increase that exceeds 0.1 mrem. Figure 30 shows the radionuclides that lead to this increase are strontium-90 and cesium-137. The results in Figure 30 also indicate that the unsaturated zone and saturated zone transport barriers eliminate any significant release from strontium-90, cesium-137, and americium-241. These barriers provide sufficient delay that these radionuclides decay away entirely in the base-case model.

In addition, these barriers also affect the release of other radionuclides. However, the effect is less dramatic than for the radionuclides whose contribution is eliminated altogether by the delay in the barriers. The natural barriers do, however, reduce the estimate of the peak mean annual dose associated with these other radionuclides by about a factor of 10 to 100. This change does not result in a mean annual dose for these other radionuclides exceeding 0.1 mrem. Accordingly, the modes for transport of these other radionuclides, including the component for transport of dissolved radionuclides and the model for transport of colloidally associated radionuclides do not play a significant role in the estimate of mean annual dose.

The individual roles of the unsaturated zone and saturated zone transport barriers are examined in Figures 31 through 34. Figures 31 and 32 show the estimate of mean annual dose when the saturated zone is included but the unsaturated zone remains neutralized. The studies in this case are conducted by discharging the release from the engineered barrier system directly to the water table (no barrier effect from the unsaturated zone) and modeling the saturated zone as in the base case. Strontium-90 and cesium-137 are included in the study.

Figure 31 compares the results of this study with the case in which both barriers are neutralized and Figure 32 shows the effect on selected radionuclides. The results show that the volcanic rock below the water table and the valley-fill alluvium provide sufficient retardation and dispersion to reduce the mean annual dose for all radionuclides. These effects attenuate the release associated with the short-lived radionuclides because the probability distribution for transport of these radionuclides in the saturated zone has only a small probability of rapid travel through this barrier. The largest part of this distribution has travel times of thousands of years and, for that fraction of travel times, there is no contribution from the short-lived radionuclides.

[7] In particular, the neutralization analyses do not indicate radionuclide or groundwater travel time through the natural barriers. Assessment of these quantities must take into account realistic representation of the characteristics of these barriers. Analyses for such conditions show a mean groundwater travel time exceeding 10,000 years and less than one chance in 100 probability of groundwater travel time less than 1,000 years (BSC 2001e, Figure 11, p. 46).

The net effect on the mean annual dose is a decrease from the case in which only the saturated zone barrier is neutralized.

Figures 33 and 34 show the analogous results for the unsaturated zone barrier. In this case, the study is conducted by using the base-case model for the unsaturated zone and discharging the calculated release from the unsaturated zone directly into the water usage volume at the accessible environment (no barrier effect from the saturated zone). Figure 33 shows a large decrease from the case in which both of these barriers are neutralized. In the model for the unsaturated zone barrier, diffusive release from the engineered barrier system is transferred directly into the rock matrix of the unsaturated zone directly below the emplacement drifts. Transport in the rock matrix is very slow, largely because of its low permeability and resulting slow movement of water that carries the radionuclides. Advective release from the engineered barrier system is discharged into the fractures of the rock below the emplacement drift. Transport of radionuclides in the fractures is affected by matrix diffusion and, for radionuclides attached to colloids, filtration of colloids. These effects result in the decrease of the mean annual dose estimate shown in Figure 33. The details of these models, including the model for transferring diffusive and advective radionuclide flux from the engineered barrier system into the rock matrix, are considered to be potentially significant in that the change in the mean annual dose associated with them exceeds 0.1 mrem.

Figure 34 shows the contributions from the individual radionuclides for this study. The unsaturated zone transport barrier attenuates the estimate of mean annual dose for all radionuclides. In particular, there is no contribution from the short-lived radionuclides. Apparently the probability distribution for the travel time of these radionuclides through the unsaturated zone has such a small contribution from short travel times that the Monte Carlo sampling (300 realizations) is unable to capture any such effects. Whatever the effects might be if more realizations were included, the combined effect of these two barriers would result in such a small probability of short travel times that it is extremely unlikely that the short-lived radionuclides would travel through both of them before they decay.

The studies in this study have examined the combined roles of flow and radionuclide transport on the estimate of mean annual dose. Figures 35 and 36 show the effect when only the flow is taken into account: that is, the effect of the sorption, matrix diffusion, and colloid filtration in these barriers is removed.[8] The flow systems in the unsaturated and saturated zones are the same as in the base-case model. The results in this case are somewhat lower than those for the case where the unsaturated zone and saturated zone are neutralized entirely; however, the difference is small. Figure 36 shows that the release from strontium-90 and cesium-137 is reduced somewhat, indicating some delay due to the groundwater travel time. However, in general the effect of the flow systems alone does not appear to play a major role in determining the estimate of mean annual dose.

This study suggests that the most important effect of the natural barriers is due to retardation relative to the groundwater flow. This retardation arises from several effects and affects

[8] Because the sorption coefficients and colloid filtration are set to zero, there is no difference between the transport of colloid-associated species and dissolved species so that the partitioning between the concentrations in each case is not important. The lack of sensitivity of the results to the transport parameters leads to the conclusion that the partitioning coefficient is also not important.

different radionuclides in different ways. The radionuclide inventory includes a small number of radionuclides that are highly soluble and that do not sorb onto the rock as readily as the majority of radionuclides in the inventory. These radionuclides include carbon-14, technetium-99, and iodine-129. These radionuclides are modeled entirely as dissolved species with very small or no retardation factor due to sorption. At the same time, retardation of their migration in the fractures due to matrix diffusion processes is taken into account. These processes do play a role in affecting the migration of these radionuclides, as shown in Figure 36.

Another category of radionuclides is the set that sorbs readily onto minerals of the rock. This category includes neptunium-237 and the plutonium isotopes, and the model takes into account the effect of sorption on migration of these radionuclides as dissolved species. The model also accounts for sorption of these radionuclides onto colloids in the water. These colloids are retarded somewhat relative to the groundwater due to filtration processes, but the retardation in this case is much less than that associated with sorption of the radionuclides onto the immobile rock. At the same time, the model takes into account matrix diffusion of these radionuclides that affects migration of both dissolved species and the radionuclides attached to colloids that desorb into the water.

The final category of radionuclides addressed in the TSPA transport model is the set of radionuclides that are integrally bound to the colloids generated by degradation of the waste form. These are modeled as "irreversibly sorbed" to the colloids. Matrix diffusion and sorption of colloids onto the minerals of the rock are not included in either the unsaturated zone or saturated zone transport model. Retardation of these radionuclides due to filtration of the colloids during their migration is taken into account.

The net effect of these models is to reduce the mean annual dose of most radionuclides in the first 10,000 years by approximately a factor of 10 to 100. Given the peak mean annual dose estimate for the groundwater scenarios of less than 0.01 mrem, this factor is not considered to be a major contributor to the estimate. The effect on strontium-90, cesium-137, and americium-241 is more important however. In this case, the effect of the retardation is to reduce the mean annual dose from a peak value of more than 0.1 mrem to a level below 0.1 mrem. The retardation is sufficient to eliminate the potential hazard from these radionuclides. The fact that these radionuclides dominate the inventory of the emplaced waste gives special importance to this role of the natural barriers.

The results of these studies are consistent with the results using the TSPA-SR model. Analyses of the nominal scenario using the TSPA-SR show a small effect on the mean annual dose estimate for carbon-14, technetium-99, iodine-129, neptunium-237, plutonium-239, and plutonium-240, as well as other radionuclides (CRWMS M&O 2000a, Section 5.2.6, p. 5-19; Section 5.2.7, p. 5-20; Section 5.3.6, p. 5-41; and BSC 2001b, Section 3.2.9.2, p. 3-31; Section 3.2.10, p. 3-32).

3.3.11 Igneous Activity Probability Sensitivity Study

This study evaluates the role of the probability for igneous activity in the TSPA model. The estimate of expected risk includes this probability as a weighting factor on the estimate of consequences of the event. The arguments in Section 2 suggest that this component could play an important role in the estimate of mean annual dose. Analysis for the site suitability evaluation shows that the mean annual dose for the igneous activity scenarios estimated in that model is very nearly proportional to the mean probability of the event (CRWMS M&O 2000a, Section 5.5.2.9.1, p. 5-22). In order to see the effect for the TSPA model used in this study, this study focuses on the igneous activity eruptive release scenario that dominates the estimate of total mean annual dose.

Figure 37 compares the base-case result (mean annual probability of 1.5e-8 in the primary area of the repository) with the result using the 95th percentile value (annual probability of 4.8e-8). The resulting mean annual dose scales by a number that is very close to the ratio of these two values, confirming the result in the site suitability evaluation that the estimate of mean annual dose is essentially proportional to the mean probability in the TSPA model. It is unlikely that the probability of igneous activity would be determined to be so large as to result in a mean annual dose estimate comparable to the 15 mrem standard. However, in view of the fact that the probability could be larger and that other effects affecting the igneous activity eruptive release mean annual dose could also contribute, this TSPA model component is considered to play a potentially significant role in the estimate of expected risk.

The results of this study are consistent with the results using the TSPA-SR model. Analyses of the igneous activity scenarios using the TSPA-SR show a potentially significant effect of the probability of the event on the estimate of mean annual dose (CRWMS M&O 2000a, Section 5.2.9.1, p. 5-22).

3.3.12 Damage to Engineered Barriers by Igneous Activity Sensitivity Study

This study assesses the role in the TSPA model of the components describing damage to the engineered barriers as a result of igneous activity. These components address the effects of igneous intrusion of magma into the repository and volcanic eruption through the repository.

The study addresses the component describing damage to waste packages, drip shields, and CSNF cladding during igneous intrusion into the repository. The TSPA model identifies two zones where waste packages and drip shields are damaged during such an event. Zone 1 defines the region in which all of the waste package and drip shields are sufficiently damaged that they offer no protection for the waste, and the CSNF cladding is breached in the modeling of this event. The extent of Zone 1 is determined in the model by sampling from a probability distribution. Zone 1 contains an average of about 300 waste packages (and drip shields and cladding associated with those waste packages) in the base-case model. The TSPA model also assumes that some of the waste packages outside Zone 1 are also damaged and defines a Zone 2 in which all of the drip shields and cladding are failed, but the waste packages are only partially damaged (endcaps are failed). Zone 2 contains an average of about 2,100 waste packages in the base-case model.

Figure 38 shows the results for the igneous activity groundwater release scenario in the base-case model. This figure shows the mean annual dose and the expected contribution to this estimate from the Zone 1 and Zone 2 releases in this base-case model. These results show that that the Zone 1 releases dominate the estimate of mean annual dose in the base-case model. The sensitivity study is therefore extended only to the Zone 1 waste packages. Figure 38 shows the results for a case in which the number of Zone 1 waste packages is set to the 95th percentile value, more than 900. As can be seen in the comparison of this result with the base-case result, the estimate of mean annual dose is essentially proportional to the mean number of waste package and drip shield combinations disrupted in the event. A much larger number of waste packages and drip shields disrupted could therefore result in a much higher estimate of mean annual dose than the base-case estimate. This possibility considered together with the fact that the mean annual dose estimate is also proportional to the mean probability of occurrence of igneous activity leads to the conclusion that this factor is considered to play a potentially significant role in the estimate of expected risk.

The model for the disruption to the engineered barriers by the intrusion is therefore considered to be important to the estimate of mean annual dose. This conclusion also extends to the Zone 2 waste packages. In the base-case model, the damage to the Zone 2 packages is assumed to be failure of the endcaps. However, the damage could be more extensive, e.g., waste package degradation rates could be changed due to changes in the environments in the drifts and waste packages could fail much earlier in Zone 2 than in the nominal scenario. Accordingly, the conclusion about the importance of the damage to the engineered barriers applies to both Zone 1 and Zone 2 waste packages.

This study also considers the role in the TSPA model of the damage caused during an igneous eruption. The TSPA model assumes that all of the waste in waste packages intersected by an eruptive vent is erupted to the surface. The model determines the number of these waste packages from the number of vents intersecting emplacement drifts and the number of waste packages in a given drift that are intercepted by the vent. The number of waste packages so intercepted in the base-case averages about 11. The analyses for the site suitability evaluation considered the impact of setting the number of waste packages intercepted to the 95th percentile (19) and the 5th percentile (7) values (BSC 2001a, Section 14.3.3.7, p. 14-24). These results show a corresponding variation in the estimate of mean annual dose that scales according to the mean number of packages intercepted. The estimate of mean annual dose for the eruptive release scenario therefore appears to be sensitive to the amount of waste erupted, as indicated in this model by the number of waste packages intersected by an eruptive vent.

These results are consistent with the results using the TSPA-SR model. Analyses of the igneous activity scenarios using the TSPA-SR show a direct dependence of the estimate of mean annual dose on the amount of waste erupted as indicated by the number of waste packages intersected (CRWMS M&O 2000a, Section 5.2.9.6, p. 5-24; Section 5.2.9.7, p. 5-24).

3.3.13 Atmospheric Transport of Erupted Radionuclides Sensitivity Studies

These studies evaluate the role of various factors in the model for transport of radionuclides in the atmosphere following an eruption through the repository. The first study considers the volume of tephra in the eruption into which the radionuclides are dispersed. A larger volume may result in greater dilution of the radionuclide concentrations. A larger volume may also be correlated with a greater height of eruption and therefore the potential for transport of tephra to greater distances. The previous study (see Section 3.3.12) examined the sensitivity to the mass of radionuclides erupted. This study examines only the sensitivity to the volume of erupted tephra without a correlated increase in the amount of waste erupted.

In the base-case model the volume of erupted tephra is sampled from a distribution between 0.002 km^3 and 0.44 km^3 with a mean erupted volume of 0.08 km^3. Figure 39 compares the base-case mean annual dose with that calculated with the value of the eruptive volume set to the 95th percentile value, 0.34 km^3. While this range does not go outside that of the base-case distribution, it is apparent that these results indicate that the erupted volume considered in the TSPA model does not play a strong role in determining the estimate of mean annual dose.

These results are consistent with the results using the TSPA-SR model. Analyses of the igneous activity eruptive release scenario using the TSPA-SR also do not show a significant effect of the eruptive volume on the estimate of mean annual dose (CRWMS M&O 2000a, Section 5.2.9.6, p. 5-24; Section 5.2.9.5, p. 5-24).

The second study examines the size of particles in the plume of erupted tephra. This factor affects the dispersion of the plume as the particles are transported downwind. Figure 40 evaluates the sensitivity of the estimate of mean annual dose for this scenario to the representation of particle size.[9] This figure compares the result of the base-case distribution for the size of the particles with two extreme cases for the mean and standard deviation of the particle size probability distribution. In one case, the mean and standard deviation are each set to their 95th percentile values and, in the other, they are each set to their 5th percentile values. The results show no significant impact on the estimate of mean annual dose. Accordingly, the particular particle size chosen does not play an important role in determining the estimate of mean annual dose.

The third study examines the role of the treatment of the wind speed and direction in the TSPA model. These factors determine the atmospheric dispersion of the radionuclide-bearing tephra plume and the resulting distribution of tephra deposited in Amargosa Valley. The available data for wind speed and direction near the Yucca Mountain site are winds measured over a period of 7 years from altitudes between 1,500 m and 5,000 m. This study considers the wind speed and the wind direction separately.

The base-case model utilizes a distribution for wind speed developed from the available data that has 5th, 50th, and 95th percentiles of about 270, 1,000, and 2,200 cm/sec, respectively (BSC 2001a, Table 14.3.3.5-1, p. 14T-4). The mean of the distribution is about 1,100 cm/sec. Figure 41 compares the base-case mean annual dose estimated for the full range of this distribution with the results fixing the wind speed at 2,200 cm/sec. These results suggest a nearly linear relation between the mean annual dose and the mean wind speed. Thus, the mean annual dose estimate depends not so much on the distribution of wind speed (e.g., with altitude) but on the average wind speed. This fact, in combination with other considerations (e.g., probability of igneous activity) leads to the conclusion that the representation of wind speed could play an important role in the estimate of mean annual dose.

Although the available data do show wind directions over the full 360 degrees, the base-case model assumes a conservative representation in which the wind is only to the south. Figure 42 compares the base-case results with those in which the full distribution of wind directions is

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[9] Particle size is also a consideration in the estimate of the BDCFs in the biosphere model. However, the BDCFs are held fixed in this analysis so that the evaluation here is only of the role of the particle size in the atmospheric transport.

utilized. This comparison shows a reduction in the mean annual dose estimate of about a factor of 20. This large effect suggests that this factor could be potentially important to the estimate of mean annual dose.

These results are consistent with the results using the TSPA-SR model. Analyses of the igneous activity eruptive release scenario using the TSPA-SR show an effect of the wind speed and direction modeled on the estimate of mean annual dose comparable to that shown here (CRWMS M&O 2000a, Section 5.2.9.2, p. 5-22; Section 5.2.9.3, p. 5-23).

3.3.14 Biosphere Characteristics Sensitivity Studies

These studies examine the importance of the biosphere component of the TSPA model. This component is manifested in the TSPA model in terms of biosphere dose conversion factors (BDCFs) that are used to translate radionuclide concentrations in groundwater and soil into annual dose. In addition, processes that affect these BDCFs, including transport processes in the biosphere, are addressed in these studies.

Figure 43 shows the sensitivity of the estimate of mean annual dose considering only the range of uncertainty in the BDCFs. The base-case result is compared with studies in which the probability distributions for all factors in the biosphere model are set to their 95th percentile values. The figure also shows the result when all these factors are set to their 5th percentile values. The estimate of mean annual dose for the igneous activity eruptive release scenario is proportional to the mean BDCFs for this scenario and shows a large change because the range of these mean values is very wide. The BDCFs in this case are considered to play a potentially significant role in the estimate of expected risk.

The change in the estimate of mean annual dose for the groundwater scenarios shows a much smaller effect because the range of uncertainty in the associated BDCFs is small. The results are proportional to the mean value of these BDCFs, as in the eruptive release case; however, it is unlikely that the models would change sufficiently to result in a mean annual dose exceeding 0.1 mrem for these scenarios. Accordingly, the BDCFs for the groundwater scenarios are not considered to play a significant role in the estimate of mean annual dose.

The results in Figure 43 are consistent with the results using the TSPA-SR model. Analyses using the TSPA-SR show results similar to those shown for this base-case model. They show, first, that the mean annual dose estimate is proportional to the BDCFs and, second, that over the range of uncertainty accounted for in the model, the variation in the estimate of mean annual dose is small (CRWMS M&O 2000a, Section 5.2.8.1, p. 5-20).

A particular factor taken into account in estimating the BDCFs for the igneous activity eruptive release scenario is the thickness of the layer of deposited tephra from the eruption through the repository. The thickness of contaminated tephra can potentially affect the BDCF used to estimate annual dose from the concentrations of radionuclides in the entire soil layer.

The importance of the tephra deposit thickness to the estimate of mean annual dose for this scenario is shown in Figure 44. This figure compares the result for the base-case model with studies for two different tephra deposit thicknesses. BDCFs for different soil thicknesses are not taken into account in the base-case model; however they were considered in the SSPA evaluations (BSC 2001a, Section 13.4.3.2, p. 1361). Accordingly, the sensitivity studies utilize the SSPA BDCFs to evaluate the role of the soil thickness. These studies consider thicknesses of 15 cm and 1 cm, respectively. The difference between these two cases amounts to a factor of less than 15 because soil removal effects reduce the overall effect of the original soil thickness. As a result, the difference in the estimate of mean annual dose is on the order of a factor of three.

The results in Figure 44 suggest that the decrease in the thickness with time due to soil erosion may not play an important role in the estimate. This role is explicitly evaluated in Figure 45. This figure shows the sensitivity of the estimate of mean annual dose to the soil removal rate. The base-case model considers a range of soil removal rates, and the base-case result is compared with the results using the 95th (0.079 cm/year) and 5th (0.061 cm/year) percentile values of the base-case range. The small variation reflects, in part, the narrow range of the uncertainty band for the removal rate.

Although soil removal addresses some aspects of soil redistribution, it does not address all of them. For example, accumulation of radionuclides due to aeolian or fluvial transport of eroded tephra is not accounted for in the base-case TSPA model. As a result, an explicit study of this effect has not yet been conducted. The possible magnitude of the effect is suggested in the comparison between the results for full 360-degree distribution of the wind in atmospheric transport with those for an assumed south-only wind direction, shown in Figure 42. In this case, the treatment amounted to a factor of 20. It is unlikely that soil redistribution would result in as large a factor. Nevertheless, until explicit analyses show otherwise, soil redistribution should be considered to have the potential to affect mean annual dose to some degree.

3.3.15 Radionuclide Inventory Sensitivity Study

This study examines the role of the representation of radionuclide inventories in the TSPA model. The base-case model considers up to 26 radionuclides.[10] Other possible radionuclides are excluded on the basis of half-life or elemental solubility and sorption characteristics.

Figure 5(a) shows the radionuclides that dominate the estimate of mean annual dose for eruptive release scenario. The radionuclides that dominate this estimate include americium-241, plutonium-238, plutonium-239, and plutonium-240. The mean annual dose estimate for each of these radionuclides is directly proportional to its inventory. The inventories for other radionuclides are not likely to play a significant role in the estimate.

Figure 5 also shows the radionuclides that dominate the estimate of mean annual dose for the groundwater release scenarios. These radionuclides include relatively soluble radionuclides, carbon-14 and technetium-99, and less soluble radionuclides, neptunium-237 and the plutonium isotopes. The estimate of mean annual dose for the soluble radionuclides depends directly on their inventories. The estimate for the less soluble radionuclides is controlled by their solubility

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[10] Although 26 radionuclides are included in the models for the engineered barrier system and for the saturated zone radionuclide transport model, only 24 are included in the base-case model for unsaturated zone radionuclide transport because of calculational limitations. That is, two short-lived radionuclides are eliminated on the grounds that their travel time would result in negligible release. These short-lived radionuclides are taken into account in the analyses of radionuclide transport in the unsaturated zone (see Section 3.3.10).

limit rather than their inventories. Inventories of the other 21 radionuclides included in the studies are not likely to play a significant role in the estimate.

These studies have not considered any of the radionuclides that have been screened from the TSPA model considerations. For example, cesium-135 and tin-136 have not been included in the studies on the basis of general screening considerations. Although these radionuclides are unlikely to contribute to annual dose, no additional insight can be provided into their role in the estimate of mean annual dose because they have not been explicitly included in these studies.

3.4 COMBINED EFFECTS SENSITIVITY STUDY

Combinations of effects from different TSPA model components are included in the Monte Carlo sampling of the one-off sensitivity studies in Section 3.3. That is, the sampling is over the ranges of parameters of the model components. For example, combinations of extreme parameters for all model components are taken into account (weighted by their probability of occurrence) in the base-case estimates. Likewise, each of the one-off sensitivity studies is probabilistic. That is, although the parameters for the model component of interest are fixed at specified values, the parameters for the other model components are sampled over their entire ranges. Accordingly, combinations of extreme parameters are taken into account to a degree in the one-off studies. What these studies do not capture, however, is the importance of simultaneous challenges to the probability models for the various processes. That is, the sensitivity studies test the probability model for one component while maintaining the probability models for the other as in the base-case TSPA study. The sensitivity studies do not therefore test whether two model components shown to play a minor role in an individual sense, even in an extreme or unrealistic representation, might play a more important role when their combined effects are considered.

Accordingly, this section reports the results of a study that explicitly considers combined effects of large changes to the probability models for several TSPA model components for the nominal and igneous activity groundwater release scenarios. The significance of the effects is assessed in terms of changes in the estimate of mean annual dose in comparison with the individual protection limit of 15 mrem. In this case, a change of 1 mrem or more in the estimate of 10,000-year mean annual dose indicates a potentially significant effect.

The following changes are implemented:

- **Climate, net infiltration, and unsaturated zone flow**—These components are changed to the representation described in Section 3.3.1. In this case the flow field associated with a mean infiltration rate of about 12 mm/year is changed to a flow field associated with a mean net infiltration rate of about 150 mm/year. Accordingly, the percolation flux is changed from the base-case flux to the much higher percolation flux associated with the changed flow field.

- **Seepage into emplacement drifts**—This component is changed so that the flux over each waste package location is increased by a factor of 10 over the value estimated in the base-case model. That is, whereas the base-case model has an average seepage flux of about 0.1 m^3/year over each waste package in areas where seepage occurs, the seepage model is changed to reflect an average flux of about 1 m^3/year in these areas.

- **In-drift chemistry**—This component is changed to lower the pH of the water in the drift invert to 4 for 100,000 years. This contrasts with the base-case model that ranges between about 5 and 7 in the first 10,000 years and between 7 and 8 thereafter.

- **Waste package and drip shield degradation**—The general corrosion rates for the waste package and drip shield are the extreme rates considered in Section 3.3.4. That is, the drip shield general corrosion rate used in this study is a fixed value that is a factor of five greater than the median rate of the base-case model, and the waste package general corrosion rate computed in the base-case model is multiplied by a factor of eight. The chemistry change does not move the degradation model into a regime where localized corrosion would be initiated; however, the extreme corrosion rates imposed here result in much earlier failure of waste packages and drip shields than in the base-case model.

- **In-package chemistry**—This component is changed to lower the pH of the water in the waste package to 4 for 100,000 years. Again, this value is outside the range considered in the base-case model.

- **CSNF cladding performance**—All cladding is assumed to be breached at the time of emplacement.

- **Dissolved radionuclide concentration limits**—The TSPA model for these limits computes a value based on the water chemistry. These limits are therefore changed because of the changes to the in-package and in-drift chemistry assumed for this study. In addition to this change, the concentration limits are further increased by a factor of 10. The result for plutonium, for example, is a net increase of several orders of magnitude.

- **Colloid-associated radionuclide concentrations**—The change in this case is the same as that considered in Section 3.3.8. That is, the concentration of colloid-associated plutonium and americium is increased to a value that is a factor of 1.5 above the maximum allowed in the base-case model.

- **Unsaturated zone radionuclide transport**—The change in this case is to force all of the radionuclides discharged from the engineered system into fractures. That is, diffusive release, rather than migrating directly into the rock matrix, goes into the fractures along with the advective release.

The results of these changes are shown in Figure 46. These changes result in a mean annual dose, conditioned on those changes, that is nearly two orders of magnitude above the estimate of mean annual dose from the base-case model in the first 10,000 years and a much larger factor after 10,000 years. The explanation for these changes can be seen in part in Figure 47 where the contributions of the individual radionuclides can be seen. The effect on individual radionuclides can be evaluated by comparing these results to the results for the corresponding scenarios in Figure 5.

The largest contributors to the igneous activity groundwater release are plutonium-239 and plutonium-240. The mean annual dose has increased in this scenario primarily as a result of three changes. First, the water chemistry is more acidic than in the base case. Figure 16 shows that this change substantially increases the solubility of this radionuclide. In addition, the study enhances this solubility by an additional factor, resulting in a concentration of this radionuclide much higher than in the base case. Finally, the amount of seepage contacting the waste is much higher than in the base case. Accordingly, if all three of these effects were to occur, the mean annual dose could approach 1 mrem, more than a factor of 10 below the individual protection standard.

The results for the nominal scenario also show these effects: the mean annual dose associated with the solubility-limited radionuclides (e.g., plutonium-239, plutonium-240, and neptunium-237) are increased due to the increase in their solubilities and, after the drip shields have failed, the increased seepage rate.

Figure 46 shows effects of other changes. The dominant contributors in the first few thousand years are the mobile radionuclides (e.g., carbon-14 and technetium-99). The neutralization of the cladding increases the mean annual dose in this case by about a factor of 10. The other change affecting these radionuclides in this period is the change to the unsaturated zone radionuclide transport model. In this case, all of the release is discharged into the fractures. Since the drip shields are still intact and diverting water away from the waste, the release from the breached waste packages is only diffusive. In the base-case model, this release goes into the rock matrix below the drifts and moves very slowly down to the water table. In this model, all of the release goes into the fractures where the radionuclides move more rapidly to the water table. The net effect of these two changes (cladding model and unsaturated zone transport model) is to enhance the mean annual dose by about 0.01 mrem.

The study for the nominal scenario shows the effect of the enhanced failure rate for the waste package and drip shield assumed in the study. The waste packages begin to fail before 10,000 years in this case and the mean annual dose for all radionuclides begins to increase accordingly. The drip shields fail before 30,000 years in this study and permit an increased amount of water to contact the waste, enhancing the release of the solubility-limited radionuclides.

This combined effects study leads to two observations. First, if all of the models were changed in the way described, the mean annual dose estimate would increase by less than 1 mrem in the first 10,000 years. That is, if the water chemistry were to be different, the solubility model were to be changed, the seepage rate were to be increased, the transport model were to be modified, and the cladding, waste package, and drip shield were all to degrade in this extreme fashion, the resulting mean annual dose estimate would still be more than an order of magnitude below the individual protection limit (15 mrem).

The second observation is that the changes seen in the combined effects study are not significantly different from the changes observed in the one-off studies. That is, the estimates of mean annual dose for the igneous activity groundwater release scenario and for the nominal scenario before the waste packages and drip shields begin to degrade are proportional to the product of the associated solubility limits and seepage rates. After the waste packages begin to degrade, the nominal scenario mean annual dose estimate increases in proportion to the number of waste packages that have breached. After the drip shields have breached, the nominal scenario increases by precisely the contribution from advective release from the waste packages. These observations are the same as those seen in the one-off sensitivity studies. Nonlinear synergistic effects are not observed, and the roles of the various model components observed in this combined effects study are the same as those indicated in the one-off studies. Accordingly, the conclusions regarding the relative priorities of these model components are not different from those inferred from the one-off studies. More importantly, nonlinear effects appear to be minor; consequently, this combined effects study suggests that the TSPA model can be explained and understood more easily than if such effects were to be important.

3.5 PRIORITIZATION OF TSPA MODEL COMPONENTS

The sensitivity studies in Section 3.3 examine the role of individual TSPA model components to the estimate of mean annual dose. These studies provide insights into the particular roles of these components in the estimate of mean annual dose. As such, the results of these studies can be used to inform the prioritization of these components with respect to their importance in showing whether the postclosure performance objectives would be met.

The results of these sensitivity studies are compared with the results of sensitivity studies conducted using the TSPA-SR model. In virtually all cases, the conclusions regarding the role of the respective TSPA model components are the same. This fact suggests that the conclusions are relatively robust and indicative of the conclusions that would be drawn even if the TSPA model used for the license application were to be somewhat different from either the base-case model used in these studies or the TSPA-SR model.

Table 2 summarizes the results of the sensitivity analyses. This table shows the relative importance of the TSPA model components to the estimate of mean annual dose for the igneous activity eruptive release scenario. The peak mean annual dose for this scenario is about 0.1 mrem. TSPA model components potentially significant to this estimate include the following:

- The probability of igneous eruption in the repository
- The amount of waste erupted
- The BDCFs associated with this scenario.

The analyses in Section 3.3.13 note that mean wind speed and direction also play some role, potentially affecting the estimate of mean annual dose by a factor of 2 to10. Likewise, the discussion in Section 3.3.14 indicates that aeolian or fluvial soil redistribution following the event could result in a similar increase. Although these effects are not significant in themselves (i.e., do not result in a mean annual dose exceeding the individual protection limit of 15 mrem), they could be important in combination. Consequently, these effects are identified as potentially significant to expected risk. Other factors included in the TSPA model, those accounted for in the model for transport of radionuclides to Amargosa Valley and soil thickness and removal after deposition of the tephra do not play an important role in the estimate (see Sections 3.3.13 and 3.3.14).

Table 2 indicates the relative importance of the TSPA model components in determining the estimate of mean annual dose for the igneous activity groundwater release scenario. Many of these components play a minor or insignificant role. The peak mean annual dose for this scenario is more than three orders of magnitude below the individual protection limit of 15 mrem and even the most important of these model components only play a potentially important role in combination with each other. The sensitivity studies indicate that each of the following model components, in combination with other components, may be potentially important to the estimate of mean annual dose for this scenario:

- Probability of intrusion of magma into the repository scenario (Section 3.3.11)
- Damage to the engineered barriers from this intrusion for the igneous activity groundwater release scenario (Section 3.3.12)
- Dissolved plutonium-239 and plutonium-240 concentrations (Section 3.3.7)
- Transport of strontium-90 and cesium-137 through the unsaturated zone and saturated zone (Section 3.3.10)
- BDCFs for the groundwater release scenarios (Section 3.3.14).

Other components, included in the TSPA model to ensure its consistency with basic physical principles, do not play a significant role in the estimate of mean annual dose for this scenario. These factors include

- **Climate and net infiltration**—This factor is represented in the TSPA model in order to define the boundary condition for flow in the unsaturated zone. This flow is taken into account in determining the seepage into the emplacement drifts and transport of radionuclides released from breached waste packages from the engineered barriers to the water table. The sensitivity study in Section 3.3.1 indicates that the details of this component are not important to the estimate of mean annual dose.
- **Seepage into emplacement drifts**—The sensitivity study in Section 3.3.2 indicates that the details of the seepage model play a small role in the estimate of mean annual dose.
- **In-drift temperature, moisture, and water chemistry**—Section 3.3.3 discusses the role of these factors in the estimate of mean annual dose. That discussion indicates that their effects on the drift invert have a negligible effect on the estimate of mean annual dose. The effects of temperature, moisture, and chemistry on waste package and drip shield degradation are not evaluated here because these effects are not explicitly taken into account in the TSPA model. The uncertainty ranges of the waste package and drift shield degradation models implicitly address these effects, and studies in Section 3.3.4 address these ranges. Studies for degradation rates that are beyond the current ranges of uncertainty do not result in significant effects on the mean annual dose estimates. These results suggest that the effects of temperature, moisture, and water chemistry on the drip shields or waste packages may not be significant. However, ongoing work on localized corrosion and temperature dependence of the waste package general corrosion rate could

result in an explicit dependence on the in-drift environments that is important. However, these environments do not have a significant effect on meeting the postclosure performance objectives in the base-case model used in these studies.

- **In-package temperature, moisture, and water chemistry**—These factors are discussed in Section 3.3.5. Their most important effect is on the dissolved radionuclide concentration limits, which are identified as important to the estimate of mean annual dose. However, over the wide range of chemical conditions evaluated in Section 3.3.5, the effect on mean annual dose is small. Considering the conditions investigated, it is unlikely that additional information would result in conditions outside this range; therefore, this conclusion is likely to be maintained.

- **Waste form degradation (including cladding degradation)**—The sensitivity studies in Section 3.3.6 show only minor dependence of the estimate of mean annual dose on cladding performance. In part, this result is due to the fact that nearly 40 percent of the waste packages contain waste that is not clad (e.g., HLW) and details of the cladding performance model have no impact on the release of this fraction of the waste. In addition, the role of the cladding is to limit exposure of the surface of the CSNF matrix to water in breached waste packages; however, this limitation only plays a minor role in affecting radionuclides such as neptunium-237 and the plutonium isotopes whose release is solubility limited rather than waste form limited. The studies also show no significant dependence on the waste form dissolution rate because this rate is so high in the base-case model.

- **Dissolved radionuclide concentrations of radionuclides other than plutonium-239 and plutonium-240**—Studies in Section 3.3.7 indicate that the concentrations of radionuclides other than plutonium-239 and plutonim-240 play only a minor estimate of mean annual dose. Radionuclides whose release rate is not controlled by their solubility, but by the waste form degradation rate, do not result in an estimate of mean annual dose that is significant in the sensitivity studies. Likewise, of the radionuclides whose release rate is limited by their solubility, only the solubility limit of plutonium has a moderately important effect on this estimate.

- **Colloid-associated radionuclide concentrations**—Studies in Section 3.3.8 indicate that these concentrations contribute little to the estimate of mean annual dose. The concentrations of radionuclides bound to waste form colloids ("irreversibly-sorbed" colloidal radionuclides) are very small and large increases in the concentrations do not show a significant impact. Radionuclides that sorb to colloids are strongly retarded in their migration due to sorption/desorption processes during migration and due to matrix diffusion. Consequently, variation of the concentrations of these radionuclides also has negligible effect on the estimate of mean annual dose.

- **Engineered barrier system radionuclide transport**—Sensitivity studies in Section 3.3.9 show negligible impact of the advective and diffusive transport characteristics of the drift invert. In part, this is due to the particular values for the invert diffusivity in the base-case TSPA model. If a lower diffusivity could be defended, the drift invert might be shown to be a more important barrier, particularly in the event the drip shield over a breached waste package remains intact and prevents flow and advective transport of the radionuclides.

- **Unsaturated zone flow and saturated zone flow**—These factors are represented by components of the TSPA model in order to describe the flow of water though the system. The flow in the unsaturated zone determines the seepage into the emplacement drifts and transport of radionuclides released from breached waste packages from the engineered barriers to the water table. The flow in the saturated zone below the water table determines the transport of radionuclides from the repository location to the accessible environment in Amargosa Valley. Sensitivity studies in Section 3.3.10 indicate that the details of these components are not important to the estimate of mean annual dose.

- **Transport of radionuclides other than strontium-90 and cesium-137 in the unsaturated zone and the saturated zone**—Studies in Section 3.3.10 show only a minor effect of unsaturated zone and saturated zone flow on the estimate of mean annual dose. The effect of transport characteristics (e.g., sorption, matrix diffusion) on relatively long-lived radionuclides, such as technetium-99, neptunium-237, plutonium-239, and plutonium-240, results in a factor of 10 reduction in the estimate of mean annual dose, and this is not considered to play a major role in determining whether the postclosure performance objectives would be met.

Table 2 indicates the relative importance of the TSPA model components to the estimate of mean annual dose for the nominal scenario. The peak of this estimate in this case is less than 0.0001 mrem and more than five orders of magnitude below the individual protection limit of 15 mrem. The study presented in Section 3.3.4 indicates that waste package degradation could significantly affect this estimate; consequently, this model component is indicated in Table 2 to be potentially significant to expected risk.

The studies in Section 3.3 indicate that no reasonable changes in any of the other model components, individually or in combination, would lead to a mean annual dose estimate exceeding 1 mrem. Therefore, none of them is regarded as significant to this estimate.

The insights summarized here are based on the results of the TSPA model used to represent the base case for these studies. In particular, they are based in part on results in which waste packages and drip shields remain substantially intact under expected conditions for more than 10,000 years. Additional work is needed to consider more aggressive environments than those in the base-case model. In this case, the degradation model may be changed to include an increased probability of failure of the waste packages before 10,000 years due to temperature-dependence of the corrosion rate or localized corrosion. Because the studies also consider the igneous activity groundwater release scenario in which a substantial number of waste packages and drip shields are disrupted before 10,000 years, such a change in the degradation model would generally not be expected to affect the conclusions regarding the roles of the various TSPA model components in the estimate of postclosure performance. A difference in these conclusions could be the role of the in-drift temperature, moisture, and water chemistry on the waste package and drip shield. If the waste package and drip shield degradation models are changed, the importance of these environments may need to be revisited.

The analyses in Section 3.3 do not address inadvertent human intrusion. In particular, they do not include explicit evaluation of the role of various TSPA model components in evaluating mean annual dose in the event exploratory drilling results in penetration of a waste package and subsequent release of radionuclides down the borehole to the water table. Analyses for this scenario were conducted for the site suitability evaluation (CRWMS M&O 2000a, Section 4.4). These analyses result in an estimated mean annual dose that is less than 0.01 mrem (CRWMS M&O 2000a, Figure 4.4-11, p. F4-47). Therefore, the factors that determine this result do not play a significant role in the estimate of mean annual dose. Consequently, the conclusions regarding the relative priority of the TSPA model components would most likely not be different if the scenario for inadvertent human intrusion had been explicitly considered in Section 3.3.

An additional consideration that could affect these conclusions is seismic activity. The DOE is currently re-evaluating the potential effect of a disruptive event in which seismic activity damages the waste packages and drip shields due to rockfall and vibratory ground motion. The considerations so far do not yet preclude combinations of event probability and degree of damage to the engineered barriers resulting in releases comparable to those from the igneous activity groundwater release scenario. Accordingly, future considerations may need to take into account groundwater release of radionuclides in a seismic activity scenario.

It is not likely that the addition of a seismic activity groundwater release scenario would affect the understanding of the roles of the various model components. TSPA model components are likely to play much the same role for a seismic activity groundwater release scenario as for the igneous activity groundwater release scenario. The most important additional considerations would be the probability of this particular scenario and the associated degree of damage to the engineered barriers. The role of these factors for this scenario is expected to be similar to the analogous factors for the igneous activity groundwater release scenario. If this scenario is determined to be an important contributor to mean annual dose, studies can be conducted to confirm this view.

Two additional points should be noted. First, the results of the sensitivity studies conducted with the base-case model considered in this report are consistent with the qualitative discussion of the waste isolation characteristics of a Yucca Mountain repository in Section 2.1. Furthermore, the results are consistent with those obtained in sensitivity studies conducted with the TSPA-SR model. These separate considerations are not independent of one another; nevertheless, they suggest an increasing convergence in the understanding of the roles of the various components of the TSPA model. While it is possible there could still be some refining of that understanding, for the most part it is likely that the basic considerations and priorities will be preserved in the future.

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4. RISK INSIGHTS FOR MODEL VALIDATION

AP-SIII.10Q, *Models*, requires that TSPA model components be validated for their intended purpose and stated limitations, and to the level of confidence required by the component's relative importance to the potential performance of the repository system[11]. The intended purpose of each model component and level of confidence required for it depends upon the licensing strategy to address the individual and groundwater protection requirements of 10 CFR 63.113. The sensitivity studies in Section 3 can be used to inform the licensing strategy regarding these considerations.

This section summarizes the relative importance of these model components to potential repository system performance indicated by the sensitivity studies reported in Section 3. This information provides the risk insights to inform the choices regarding the focus of the validation activities and level of validation needed for those TSPA model components. Table 3 provides a correlation between the TSPA model components and the areas of model validation discussed in the following sections.

For the purposes of the discussion in this section, the level of validation needed for a particular model is interpreted in terms of one of three categories:

- **Level I**—The lowest level of validation. This level includes discussion of the activities conducted during development of the model to provide confidence in it and verification that a reasonable scientific or engineering approach was taken in that development.
- **Level II**—The next level of validation. This level includes the efforts for Level I and, in addition, efforts to show that the model is conservative or that model predictions are corroborated by data or observations not used in the development of the model.
- **Level III**—The highest level of validation. This level includes the efforts for Level I and corroboration of model predictions by data or observations not used in the development of the model.

For the purpose of indicating the appropriate level of model validation, a model whose variation could lead to a potentially significant effect on expected risk (e.g., a change in the estimate of mean annual dose greater than 1 mrem) should receive Level III model validation. Models whose variation could lead to moderate effect on this estimate (e.g., a change less than 1 mrem but greater than 0.1 mrem) should receive Level II model validation. Level I validation is sufficient for all other models.

4.1 CLIMATE AND NET INFILTRATION

The climate and net infiltration component defines the representation of the water percolating into the mountain in the TSPA model. This component plays a role in determining the amount of water that might contact waste, mobilize radionuclides, and carry those radionuclides away from the repository to the water table. The TSPA sensitivity studies, however, show only a limited

[11] The term "potential performance" used here is the terminology used in AP-SIII.10Q, *Models*, in the procedure related to model validation. It is synonymous with "waste isolation" as discussed in Section 2.

sensitivity to the amount of water contacting the waste or the flow carrying radionuclides. In particular, the studies discussed in Section 3.3.1 do not show a strong sensitivity of the estimate of mean annual dose to this model component. Therefore, although the TSPA model requires that net infiltration be identified to define the unsaturated zone flow system, the details of the models for climate and net infiltration do not significantly affect the estimated performance of the repository system. Accordingly, the intended purpose of these models is to provide consistency with basic physical principles, such as conservation of mass. Validation efforts should therefore focus on this consistency. Considering the low level of importance of the model to the estimate of mean annual dose, Level I validation is appropriate.

4.2 UNSATURATED ZONE FLOW

This component describes the representation of flow in the unsaturated zone above and below the repository in the TSPA model. It defines the percolation flux at the repository horizon. This component also encompasses any effects of heat on the unsaturated zone flow accounted for in the model. As indicated in Section 4.1, the TSPA sensitivity studies do not show a strong sensitivity of the estimate of mean annual dose to any of these factors. Accordingly, the details of the associated models, including the effects of heat, do not significantly affect the estimated performance of the repository. The intended purpose of these models is therefore to provide consistency with basic physical principles, including conservation of mass and energy. Accordingly, validation efforts should focus on this consistency. Considering the low level of importance of the model to the estimate of mean annual dose, Level I model validation is appropriate.

4.3 SEEPAGE INTO EMPLACEMENT DRIFTS

This component provides the representation of water movement into the emplacement drifts from the host rock in the TSPA model. As indicated in Section 3.3.2, the TSPA sensitivity studies conducted here do not show a significant sensitivity of the estimate of mean annual dose in the first 10,000 years on the amount of seepage. Accordingly, the details of the seepage model, including the effects of heat and drift degradation, are not important to potential performance of the repository system. The purpose of the TSPA model component for seepage is therefore to ensure appropriate representation of basic physical principles such as conservation of mass and energy. Validation efforts should therefore focus on this representation. Considering the low level of importance of the model to the estimate of mean annual dose, Level I model validation is appropriate.

4.4 IN-DRIFT MOISTURE AND CHEMISTRY

This component provides the representation of the moisture and water chemistry conditions in the drift invert in the TSPA model. These conditions determine the radionuclide transport properties of the invert in this model. The discussion in Section 3.3.3 indicates that the estimate of mean annual dose in the first 10,000 years is not sensitive to these conditions. That is, although they are represented in the TSPA model, the details of the models used to represent them are not important to potential performance. The intended purpose of these models is therefore to provide consistency with basic physical principles such as conservation of mass and energy. Validation efforts should therefore focus on this consistency. Considering the low level of importance of the model to the estimate of mean annual dose, Level I validation is appropriate.

Technically speaking, this component should provide the description of the moisture and chemical conditions on the waste package and drip shield. However, these conditions are implicitly taken into account in terms of the uncertainties in the waste package and drip shield degradation rates in the current model. These degradation rates are important to the estimate of mean annual dose. Accordingly, the validation efforts should focus on confirming that the ranges accounted for in the degradation models adequately represent the conditions expected over the next 10,000 years. Considering the wide margin provided in the current degradation models, Level II validation is appropriate.

4.5 WASTE PACKAGE AND DRIP SHIELD DEGRADATION

The models in this area include corrosion of the waste package and drip shield. This area also includes early failure of these barriers, e.g., early failure of the waste package due to improper heat treatment. The TSPA sensitivity studies in Section 3.3.4 show that the performance of the waste package plays a potentially important role in the estimate of mean annual dose in the first 10,000 years. In particular, this estimate is strongly affected if there is a significant probability of waste package breaching before 10,000 years.

The base-case model does not show significant breaching of waste packages before 10,000 years, either as a result of normal degradation or as a result of early failures associated with improper heat treatment. Accordingly, validation of the current model should show that, considering the technical bases for the parameter values and ranges, probability distributions used in the process-level models, and TSPA abstractions implementing these process-level models, there is reasonable confidence in that model. For example, the current degradation model results in less than 1 percent of the waste packages failing before 10,000 years. These results reflect the following features evidenced in the current model:

- The fraction of waste packages expected to fail early due to improper heat treatment or other fabrication defects
- The general corrosion rates of the waste package outer barrier material, including enhancements of the general corrosion rate by microbial effects and aging
- Factors affecting SCC degradation (stress thresholds for crack growth, number and character of defects in the weld region, etc.)
- Localized corrosion (crevice corrosion or pitting) of the waste package outer barrier.

Validation efforts should therefore focus on these features. Considering the importance of the degradation models to the estimate of mean annual dose, Level III validation is appropriate for the waste package degradation model. Level I validation is appropriate for the drip shield degradation model. Validation of the moisture and chemical conditions that could affect these degradation rates is discussed in Section 4.4.

4.6 RADIONUCLIDE RELEASE RATES AND CONCENTRATIONS

The components in this category define the rate of release of radionuclides from the engineered barrier system. These TSPA model components include the following:

- Radionuclide inventory in each waste package
- Temperature and water in the waste package and chemistry of that water (and the evolution of those factors with time)
- Degradation of the waste form, including breaching of CSNF cladding and dissolution of the waste form matrix
- Concentrations of dissolved radionuclides and colloid-associated radionuclides
- Radionuclide transport from the waste package and through the drift invert.

The TSPA sensitivity studies in Sections 3.3.5 through 3.3.9 indicate that the estimate of mean annual dose in the first 10,000 years has only a minor dependence on in-package temperature, moisture, or chemistry, CSNF cladding degradation, waste form dissolution, colloid-associated radionuclide concentrations, or transport characteristics of the drift invert in the current TSPA model. That is, although the TSPA model requires that model components for these quantities be specified, the details of these components are not important to the quantitative estimate of postclosure system performance. The intended purpose of these components is therefore to provide consistency with basic physical principles such as conservation of mass and energy. Accordingly, an appropriate level of confidence can be achieved by demonstrating that these models each conform to generally accepted physical principles. Considering the low level of importance of these components to the estimate of mean annual dose, Level I validation is appropriate.

The mean annual dose estimate is directly proportional to the inventories of the radionuclides that dominate that estimate. The estimate for the igneous activity eruptive release scenario is less than 0.1 mrem and the radionuclides that dominate that estimate include americium-241, plutonium-238, plutonium-239, and plutonium-240. Validation of the inventories of these four radionuclides should consider their range of uncertainty and variability. Considering the level of mean annual dose associated with these radionuclides in this scenario, Level II validation of their inventories is appropriate. The estimate of the 10,000-year mean annual dose for the groundwater release scenarios is less than 0.01 mrem. In addition to plutonium-239 and plutonium-240, this estimate is dominated by carbon-14, technetium-99, iodine-129, and neptunium-237. The total contribution of these radionuclides comes to less than 0.001 mrem. Therefore, an adequate level of confidence in the inventories of these radionuclides would be obtained if the range of uncertainty and variability in their values were evaluated. Considering the mean annual dose associated with the groundwater release scenarios, a low level of validation is needed for these radionuclide inventories. Finally, the screening of radionuclides from the TSPA studies should be validated. Validation activities in this case should consider the range of characteristics (half-life, solubility, and retardation characteristics) of these radionuclides. Considering the low level of importance of these radionuclides, Level I validation of their inventories is appropriate.

Sensitivity studies show some dependence of the estimate of mean annual dose and groundwater concentrations on the solubility limit of plutonium. The current model considers a very wide range for this solubility, but an expected value that is less than 10 mg/L under expected conditions. The estimate of mean annual dose for the groundwater release scenarios is less than 0.01 mrem. An adequate degree of confidence in the plutonium solubility would be obtained if the range of uncertainties in the models and the abstractions implementing them were evaluated. Considering the level of importance of the plutonium solubility limit to the estimate of mean annual dose, Level II validation is needed. Level I validation is appropriate for the solubility of other elements.

The ratio of diffusive and advective transport through the drift invert plays a role in determining the transport of radionuclides in the unsaturated zone. Diffusive release from the drift invert is transferred to the rock matrix and advective release is transferred into the fracture system of the host rock. Validation of this ratio should consider the assumptions and parameter ranges taken into account in determining its value. In view of the low importance of the transport characteristics of the drift invert and the role of transport in the unsaturated zone, Level I validation is appropriate in this case.

4.7 UNSATURATED ZONE RADIONUCLIDE TRANSPORT

This component provides the representation of the pathways for transport of radionuclides in the unsaturated zone and the radionuclide transport characteristics along those pathways in the TSPA model. It describes the drift-scale and mountain-scale processes that disperse and delay migration of radionuclides from the engineered barrier system to the water table.

The TSPA sensitivity studies in Section 3.3.10 show a moderately important effect of the unsaturated zone radionuclide transport barrier on the estimate of mean annual dose for the groundwater release scenarios. The most important role of this barrier is the delay to the transport of radionuclides of relatively short half-life but high potential dose, including strontium-90 and cesium-137. The current model results in travel time through the unsaturated zone of several thousand years, enough time to reduce the mean annual dose for these radionuclides to negligible levels. Therefore, validation efforts should consider the uncertainties in the representation of transport characteristics in the transport model as they affect the travel time estimate for these two radionuclides. Considering the level of importance of the model to the estimate of mean annual dose, Level II validation is appropriate.

The model validation efforts should consider the different scales to which the model is applied. That is, an appropriate degree of confidence in the model would be obtained if the uncertainties at the mountain scale and at the drift scale were each explicitly evaluated. Of particular importance in this regard is the validation of the drift-scale model. This model assumes that advective release from the engineered barrier system enters only the host rock fracture system while diffusive release enters only the matrix. The validation efforts should include a focus providing confidence in this representation.

Transport characteristics of other, more-mobile radionuclides such as carbon-14, technetium-99, and iodine-129 play no significant role in the estimate of mean annual dose. TSPA sensitivity studies show that, in addition, the estimate of mean annual dose is not sensitive to the specific transport characteristics of radionuclides, such as neptunium-237 and the plutonium isotopes. Accordingly, an appropriate level of confidence would be obtained for the models of transport of these radionuclides by showing they are consistent with measured values of the transport characteristics, such as sorption and matrix diffusion. Considering the low level of importance of the model to the estimate of mean annual dose, Level I validation is appropriate.

4.8 SATURATED ZONE FLOW AND RADIONUCLIDE TRANSPORT

This component provides the TSPA representation of the flow of water below the water table that could transport radionuclides from the repository location to the accessible environment in Amargosa Valley. The saturated zone model therefore describes the pathways for transport of the radionuclides in the volcanic aquifers and the valley fill alluvium and the fluxes of water along these pathways. The model also describes the transport characteristics of the radionuclides as they move in these pathways.

The TSPA sensitivity studies in Section 3.3.10 show a moderately important effect of the combined unsaturated zone and saturated zone radionuclide transport barriers on the estimate of mean annual dose for the groundwater release scenarios. The most important role of these barriers is the delay to the transport of radionuclides of relatively short half-life but high potential dose, including strontium-90 and cesium-137. The current model results in travel time through the volcanic aquifers and the valley-fill alluvium of thousands of years, enough time to reduce the mean annual dose for these radionuclides to negligible levels. Therefore, validation efforts should consider the uncertainties in the representation of transport characteristics in the transport model as they affect the travel time estimate for these two radionuclides. Considering the level of importance of the model to the estimate of mean annual dose, Level II validation is appropriate.

4.9 PROBABILITY OF IGNEOUS ACTIVITY

The probability of igneous activity is represented in the TSPA model in terms of two factors: (1) igneous activity eruptive release probability (probability of igneous eruption through the repository and (2) igneous activity groundwater release probability (probability of igneous intrusion into emplacement drifts). The igneous activity eruptive release scenario dominates the estimates of mean annual dose in the base-case TSPA model. The peak mean annual dose is estimated to be on the order of 0.1 mrem, and this estimate is proportional to the mean probability of the eruption of the current model. Validation of this quantity should therefore consider the parameter values and ranges as they affect this probability estimate. Considering the potential importance of this estimate to mean annual dose, Level III validation is appropriate.

The estimate of mean annual dose for the igneous activity groundwater release scenario is less than 0.01 mrem in the first 10,000 years and this estimate is proportional to the mean event probability of the current model. While Level II validation is appropriate for this probability, the validation activities for the probability of igneous intrusion would be related to those for igneous eruption.

4.10 DAMAGE TO ENGINEERED BARRIERS BY IGNEOUS ACTIVITY

The TSPA model includes components to take into account (1) damage to waste packages, drip shields, and cladding as a result of igneous intrusion and (2) damage to waste packages, drip shields, and cladding as a result of igneous eruption. The eruptive release scenario dominates the mean annual dose estimate in the current model. The peak value of this estimate is on the order of 0.1 mrem, and this value is proportional to the mean amount of radionuclides erupted. In the TSPA model this quantity is determined from the mean number of waste packages intersected by conduits during the event. Adequate confidence in this number would therefore be obtained by considering parameters, assumed ranges, and bounding assumptions in the context of this quantity. Considering the importance of the mean number of waste packages disrupted to the estimate of mean annual dose, Level III validation is appropriate.

The estimate of mean annual dose for the igneous activity groundwater release scenario is less than 0.01 mrem in the first 10,000 years, and this estimate is proportional to the mean number of waste packages and drip shields disrupted by magma intruding into the emplacement drifts, i.e., those in Zone 1. The TSPA model also considers damage to waste packages outside Zone 1, but the limited extent of this damage in the model (on the order of 10 cm^2 breach area) leads to an insignificant contribution to the mean annual dose. Therefore, validation efforts focus on the estimate of the number of waste packages in Zone 1 and the degree of damage to waste packages in Zone 2. Considering the level of importance of these factors to the estimate of mean annual dose, Level II validation is appropriate.

4.11 ATMOSPHERIC TRANSPORT OF ERUPTED RADIONUCLIDES

The TSPA model includes a component to represent transport of radionuclides in the atmosphere following eruption from the repository. This component is determined by factors including the volume of erupted material, the particle size of the radionuclide-bearing tephra, the wind speed and direction, and the deposition of tephra in Amargosa Valley. The TSPA sensitivity studies in Sections 3.3.13 indicate that, of these factors, the only ones that bear significantly on the estimate of mean annual dose are the mean wind speed and direction. Accordingly, adequate confidence in this TSPA model area would be obtained by considering the uncertainties and assumptions in the representation of these factors. Considering the effect of these factors on the estimate of mean annual dose, Level II validation is appropriate.

4.12 BIOSPHERE CHARACTERISTICS

This component provides the representation of processes leading to uptake of radionuclides by individuals and the effects of that uptake in the TSPA model. The outputs of this model are groundwater release and eruptive release BDCFs that translate radionuclide concentrations in groundwater, air, and soil into annual dose. The mean annual dose is therefore directly proportional to these BDCFs.

The submodels associated with the igneous activity eruptive release scenario have a stronger influence on the estimate of total mean annual dose than those associated with the groundwater release scenarios. The mean annual dose for this scenario is currently estimated to be less than 0.1 mrem, and the contribution of the biosphere submodels to uncertainty in this estimate is less than a factor of two. The soil, air, and inhalation submodels dominate the BDCFs for the eruptive release scenario. The validation activities should therefore focus on considerations of the conceptual models, process-level models, and abstractions for the TSPA model as they apply to these submodels. In addition, the validation activities should consider the representation of soil thickness, removal, and aeolian and fluvial redistribution. Considering the importance of the biosphere models and the models for the soil thickness, removal, and redistribution to mean annual dose, Level III validation is appropriate.

The mean annual dose for the groundwater release scenarios is currently estimated to be less than 0.01 mrem in the first 10,000 years. The contribution of the biosphere submodels for this scenario to uncertainty in this estimate is less than a factor of two. The soil submodel, the plant submodel, and the ingestion submodel dominate the estimate of the BDCF for these scenarios. Level I validation is appropriate for these submodels.

5. RISK INSIGHTS FOR KTI AGREEMENTS

The NRC has identified KTIs associated with the process models that underlie the TSPA model. The NRC and the DOE agreed upon the information that would be needed to address these KTIs. The NRC determined that these agreements provide a basis for concluding that development of an acceptable license application is achievable. The sensitivity studies presented in Section 3 provide additional information that was not available at the time those KTIs agreements were made. These studies provide insights into the relative importance to the DOE safety case of the information that would be obtained in meeting the original agreements. Section 5.1 summarizes the relative importance of the various KTI agreements in the context of the conclusions regarding the prioritization of TSPA model components in Section 3. Section 5.2 provides specific examples of the way this information may be used to address the KTI agreements.

5.1 PRIORITIZATION OF TSPA MODEL AREAS AND COMPONENTS RELATED TO KTI AGREEMENTS

The information in Section 3 applies to TSPA model areas and model components. Many of the KTI agreements can be correlated directly with one or more of these model areas. Table 4 shows this correlation. The following sections discuss the sensitivity studies reported in Section 3 in the context of the associated KTI agreements in Table 4.

5.1.1 Unsaturated Zone Flow and Seepage

This TSPA model area includes the effects of climate, net infiltration, unsaturated zone flow, seepage, and mechanical effects on unsaturated zone flow and seepage. Table 4 indicates relevant KTI agreements in these areas. These areas all pertain to the water that enters the emplacement drifts and that could mobilize waste in breached waste packages and the flow that could transport radionuclides to the water table.

The TSPA sensitivity studies in Sections 3.3.1 and 3.3.2 show little sensitivity to the amount of water seeping into the drifts and the effects that determine this seepage. In particular, the studies in Section 3.3.1 do not show a strong sensitivity of the estimate of mean annual dose to this model component. They show only limited sensitivity of the estimate of mean annual dose associated with the groundwater release pathways and negligible sensitivity of the total mean annual dose estimate. Accordingly, the information that would be provided in satisfying the KTI agreements indicated for these areas in Table 4 would play little role in determining whether the individual and groundwater protection performance objectives would be met.

5.1.2 In-Drift Moisture and Chemistry

No sensitivity studies are shown in Section 3.3 that directly evaluate the importance of the amount or chemistry of the water contacting the drip shields or waste packages. However, discussions in Section 3.3.3 indicate that other information supports the conclusion that these factors do not play a significant role in determining the estimate of mean annual dose.

Effects of in-drift chemistry on waste package and drip shield degradation are investigated indirectly in terms of the range of uncertainty (that arises in part from variations in the chemistry of water contacting these elements) in these degradation rates. These studies show no impact on performance in the first 10,000 years. Effects of chemistry of water in the drift invert are evaluated directly in Section 3.3.3 and are shown to have no significant impact on the estimate of mean annual dose.

These considerations suggest that the mean annual dose estimates for 10,000 years are not sensitive to these factors. Accordingly, information that would be acquired in meeting the KTI agreements indicated in Table 4 are not likely to play a significant role in assessing the ability to meet the individual and groundwater protection requirements.

5.1.3 Waste Package and Drip Shield Degradation

The roles of waste package and drip shield degradation are investigated in the sensitivity studies in Section 3.3.4. These studies indicate that the degradation rate of the waste package is important to meeting the individual and groundwater protection requirements as they pertain to demonstrating that significant degradation of the waste package occurs before 10,000 years. Accordingly, information that would be obtained in satisfying the KTI agreements indicated in Table 4 associated with the waste package degradation rate is expected to be relevant to showing that the individual and groundwater protection requirements are met.

The studies of the drip shield performance model indicate that the degradation rate of the drip shield is not important to meeting the individual and groundwater protection requirements. Consequently, information that would be obtained in satisfying the KTI agreements that relates solely to the drip shield degradation is not expected to be important to showing that these requirements are met.

5.1.4 Mechanical Disruption of Engineered Barriers

This area includes the effects of rockfall and ground motion on the drip shields, waste packages, and CSNF cladding. None of the sensitivity studies discussed in Section 3.3 specifically address these effects. However, they are addressed indirectly in the sensitivity studies in Sections 3.3.4 (drip shield performance and waste package performance) and 3.3.6 (CSNF cladding performance). As discussed in Section 3.5, drip shield performance is not important to meeting the individual protection requirement. Consequently, the information that would be obtained from those agreements that relate solely to the effects of rockfall or ground motion on the drip shield is not expected to be important to showing that this requirement is met.

Section 3.5, however, does discuss the fact that waste package performance is important to meeting the individual and groundwater protection requirements. Accordingly, the agreements in Table 4 that focus on the range of rockfall and seismic activity that should be considered in evaluating waste package performance are considered important to the postclosure safety case.

Studies presented in Section 3.3.6 examine the sensitivity of the estimate of mean annual dose to CSNF cladding performance and show this performance does not play a significant role in this regard. Accordingly, the information that would be provided in satisfying the KTI agreements, indicated in Table 4 to be associated with cladding performance, is not expected to be important to demonstrating that the individual and groundwater protection requirements would be met.

5.1.5 Quantity and Chemistry of Water Contacting Waste Forms

This area addresses in-package temperature, moisture, and chemistry. The discussion in Section 3.3.5 indicates that none of these factors is potentially important to the estimate of mean annual dose. Therefore, none of the information to be obtained in meeting those KTI agreements indicated in Table 4 associated with these factors is important to showing that the individual and groundwater protection requirements would be met.

5.1.6 Radionuclide Release Rates and Concentrations

This area addresses waste form degradation (including CSNF degradation), dissolved and colloid-associated radionuclide concentrations, and transport of radionuclides through the engineered barrier system to the host rock. The sensitivity studies in Sections 3.3.6 through 3.3.9 indicate that only one of these factors, dissolved radionuclide concentrations, is potentially important to the estimate of mean annual dose. In particular, only the information to be obtained in meeting those KTI agreements indicated in Table 4 associated with solubility limits of neptunium and plutonium is important to showing that the individual and groundwater protection requirements would be met.

5.1.7 Unsaturated Zone Radionuclide Transport

This area includes all of the factors affecting transport of radionuclides from the emplacement drifts to the water table below the repository. It includes drift-scale effects that describe the movement of diffusive and advective releases from the engineered barrier system into the host rock and the mountain-scale transport of radionuclides down through the unsaturated zone. Table 4 indicates KTI agreements relevant to this area.

The TSPA sensitivity studies in Section 3.3.10 show little sensitivity to the transport characteristics of most radionuclides that might be released to the unsaturated zone. They do show, however, some sensitivity to the travel time of strontium, cesium, and americium through the unsaturated zone. None of the agreements pertain to the transport properties of these radionuclides. Accordingly, the information that would be provided in satisfying the KTI agreements that are given in Table 4 is expected to play no significant role in assessing whether the individual and groundwater protection requirements would be met.

5.1.8 Saturated Zone Flow and Radionuclide Transport

This area includes the flow system below the water table and the radionuclide transport pathways in that system from the repository to the accessible environment in Amargosa Valley, in particular, the pathways in the volcanic aquifers and the valley-fill alluvium. This area also includes all the factors that affect transport of radionuclides in those pathways. Table 4 indicates KTI agreements relevant to this area.

The TSPA sensitivity studies in Section 3.3.10 show little sensitivity to the characterization of flow or radionuclide pathways in the saturated zone. They also show no significant sensitivity to transport characteristics of most of the radionuclides that would be released from the unsaturated zone to the water table. The studies show a limited sensitivity to the travel time of strontium, cesium, and americium through the volcanic rocks; however, none of the agreements pertain to the transport properties of these radionuclides. Accordingly, the information that would be provided in satisfying the KTI agreements that are given in Table 4 is expected to play no significant role in assessing whether the individual and groundwater protection requirements would be met.

5.1.9 Disruption of Engineered Barriers by Igneous Activity

This area includes the probability of igneous activity that could disrupt the repository. In particular, it includes the probability of igneous intrusion into the repository and the probability of eruption through the repository. It also includes the degree of damage to the waste packages, drip shields, and cladding in these cases. Table 4 indicates the single KTI agreement associated with this area.

The TSPA studies discussed in Section 3.3.11 indicate that this probability, particularly the probability for eruptive release of radionuclides, is important to the estimate of mean annual dose. Accordingly, the information needed to support the DOE estimate of this probability is expected to be important to the license application safety case regarding the individual protection requirements.

The TSPA studies discussed in Section 3.3.12 indicate that the disruption associated with igneous eruption through the repository is important to the estimate of mean annual dose. Accordingly, the information that would be obtained regarding the number of waste packages that would be disrupted as a result of igneous eruption and the amount of waste that would be erupted is expected to be important to the license application safety case regarding the individual protection requirements.

Information regarding the disruption of engineered barriers as a result of intrusion into the repository is less important to the safety case. The mean annual dose for this case is estimated to be less than 0.01 mrem based on disruption of an average of 300 waste packages. On this basis, disruption of all waste packages would result in a mean annual dose that is less than 1 mrem, more than an order of magnitude below the individual protection limit.

5.1.10 Atmospheric Transport of Erupted Radionuclides

This area includes those factors that affect the transport of radionuclides following volcanic eruption through the repository. These factors include the volume of erupted material, the characterization of this material (e.g., particle size) and the wind speed and direction. Table 4 indicates the KTI agreements associated with these factors.

The TSPA studies discussed in Section 3.3.13 indicate that the only factors associated with atmospheric transport of the radionuclides moderately important to the estimate of mean annual dose are the wind speed and direction. None of the remaining agreements are related to these factors. Therefore, for this model area none of the remaining agreements would result in information that is important to showing the individual and groundwater protection requirements would be met.

5.1.11 Biosphere Characteristics

This area includes the model for the BDCFs for the eruptive release and groundwater release scenarios and the submodels for calculating these BDCFs. The discussion in Section 3.3.14 indicates that the estimate of mean annual dose is directly proportional to the BDCFs of the radionuclides that dominate this estimate. They are therefore important to showing the individual and groundwater protection requirements would be met. Accordingly, the information that would be developed in meeting the agreements indicated in Table 4 associated with this area is expected to be important to the postclosure safety case. Some of this information will be more important than other information. In particular, the information associated with the soil, air, and inhalation submodels is expected to be most important because these submodels dominate the estimate of the BDCFs for the eruptive release scenario. At a somewhat lower level of importance are the plant submodel and the ingestion submodel that, along with the soil submodel, dominate the estimate of the BDCFs for the groundwater release scenarios. Other submodels play a minor role in these estimates.

5.1.12 Postclosure Criticality

The sensitivity studies discussed in Section 3.3 do not address the potential for postclosure criticality. The focus of work in this area is to show that postclosure criticality is not a credible possibility. The agreements in Table 4 associated with development of this probability screening argument are therefore expected to be important to the DOE postclosure safety case. Information developed in meeting those agreements associated with evaluating consequences of postclosure criticality is currently expected to be unimportant to showing that the individual and groundwater protection requirements would be met. If the studies show that postclosure criticality cannot be screened out, consequences of such a criticality would need to be evaluated.

5.2 EXAMPLE KTI AGREEMENTS

Table 5 lists five of the KTI agreements. These are considered in this section as examples of the approach to addressing the KTI agreements. These pertain to three TSPA model areas: (1) drip shield SCC degradation, (2) climate and net infiltration, and (3) unsaturated zone flow. Each of these areas is addressed separately. These are considered here to show how the information developed in Section 3 can be used to address these agreements and to help in resolution of specific KTIs.

5.2.1 TSPAI 3.03—Drip Shield Stress Corrosion Cracking

Total System Performance Assessment and Integration (TSPAI) agreement 3.03 relates to the technical basis for crack arrest and plugging of crack openings (including the impact of oxide wedging and stress redistribution) in assessing the effects of SCC degradation of the drip shield and waste package on repository performance. The underlying issue is the representation of the crack in estimating flow of water through a crack and transport of radionuclides in this flow.

The waste package and drip shield degradation component of the TSPA model estimates the degree of breaching of the drip shields over time. This information is used to estimate the fraction of seepage incident on the drip shield that is transmitted through to the waste package and the nature of radionuclide transport out of the waste package. In the August 2001 TSPAI

Technical Exchange (Cornell 2001), the NRC indicated that the DOE needs to provide the technical basis for the tight crack geometries that prevent advective transport through cracks in the drip shield. The DOE acknowledged that the screening arguments regarding these crack openings could be strengthened. Table 5 gives the agreement between the NRC and the DOE made at that time.

The sensitivity studies discussed in Section 3.3.4 provide additional insight regarding the importance of the representation of drip shield crack geometries in determining whether the individual and groundwater protection performance objectives would be met. In particular, the studies discussed in Section 3.3.4 show that, if the drip shield is assumed to be completely breached (i.e., completely open with respect to transmission of water) at the time of emplacement, the associated increase in mean annual dose is small. Thus, it can be concluded that the particular representation of cracks in the drip shield has little effect on overall repository performance. The information regarding crack growth and plugging of cracks in the drip shield that would be developed in accordance with agreement TSPAI 3.03 is therefore not important to showing that the individual and groundwater protection requirements would be met.

5.2.2 TSPAI 3.19, USFIC 3.01, and USFIC 3.02—Climate and Net Infiltration

TSPAI agreement 3.19 and Unsaturated and Saturated Flow under Isothermal Conditions (USFIC) agreements 3.01 and 3.02 relate to the technical basis for the TSPA models for climate and net infiltration. The concern is that the net infiltration model could underestimate infiltration flux.

In the August 16-17, 2000, USFIC Technical Exchange (Reamer and Williams 2000a) and the October 31-November 2, 2000, USFIC Technical Exchange (Reamer and Williams 2000b), the NRC questioned the representativeness of the upper bounds of the infiltration model for the future climate states, e.g., glacial transition climates. The NRC concern was that the DOE models would underestimate infiltration and, therefore, underestimate mean annual dose. The DOE agreed that the technical basis for the climate and net infiltration model could be strengthened. Agreements between the NRC and the DOE in this regard are provided in Table 5.

The study in Section 3.3.1 provides additional information bearing on the NRC concern. This study examines the sensitivity of the estimate of mean annual dose to the model for net infiltration. As indicated in Section 3.3.1, precipitation at Yucca Mountain currently averages about 190 mm/year, and the model results in a mean infiltration flux of about 4.6 mm/year corresponding to these conditions. Extrapolation of the model to account for potential increases in precipitation results in a mean infiltration flux of about 12 mm/year over this period. The sensitivity analysis considers an infiltration flux that averages about 150 mm/year, nearly equal to the present-day precipitation. This range is sufficient to evaluate the role of the infiltration model in evaluating repository performance.[12] The analyses show an increase in the mean annual dose that is less than 0.001 mrem for the nominal scenario, in which waste packages and

[12] The analysis explores the difference between the base-case model and the extreme infiltration model. The issue of potentially important effects for models intermediate to these two cases is discussed in Section 3.3.1 and 3.3.2. The concern in this case is that flow focusing or episodicity effects might result in locally high fluxes that could result in increased releases. However, the analyses in Sections 3.3.1 and 3.3.2 consider a sufficiently high range of infiltration and seepage that such effects would be accounted for.

drip shields divert water. The results also show an increase in the mean annual dose that is less than 0.01 mrem for the igneous activity groundwater release scenario in which these water-diversion barriers are disrupted. The change in each case is considered to be insignificant. Consequently, the information that would be provided in satisfying the agreements associated with the infiltration model (TSPAI 3.19, USFIC 3.01, and USFIC 3.02) is not expected to play an important role in showing whether the individual and groundwater protection requirements would be met.

The sensitivity study in Section 3.4 considers the effect of variation in the infiltration in combination with other models, including the models for seepage, in-drift and in-package chemistry, dissolved and colloid-associated radionuclide concentrations, waste package and drip shield degradation, and unsaturated zone radionuclide transport. This study shows effects that are larger than those shown in Section 3.3.1. Nevertheless, the resulting change in the estimate of mean annual dose is still well below the individual protection limit. Further, when the probability of this combination of changes is taken into account to provide a risk-informed context, the overall effect is not important. Accordingly, the information that would be developed in meeting these particular agreements is not important to showing that the individual and groundwater protection requirements would be met.

5.2.3 TSPAI 3.22—Unsaturated Zone Flow

TSPAI agreement 3.22 relates to the assumption that an unsaturated zone flow model calibrated to present-day conditions can be used to forecast the flow for future climate states. In the August 2001 TSPAI Technical Exchange (Reamer and Gil 2001), the NRC questioned the representativeness of the evapotranspiration model in predicting conditions during future climates and the appropriateness of using a hydrologic property set from a calibrated model on current climate conditions to forecast flow for future climate conditions. The DOE agreed that additional information would strengthen the basis for the extrapolation of the unsaturated zone flow model to future conditions. The resulting agreement between the DOE and the NRC on this matter is provided in Table 5.

The sensitivity study in Section 3.3.1, and that is summarized in Section 5.2.2, provides additional insight into this issue. This study considers the effect of using a much different flow field than the base-case model. The flow field in this case is one associated with the glacial maximum climate and results in much greater percolation flux than for the current or glacial transition climate. The impact on the estimate of mean annual dose is not significant. This result suggests that the issue of extrapolation of the current unsaturated zone flow model to future climate conditions is not important to the determination of whether the individual and groundwater protection performance objectives would be met and that the information that would be provided in meeting TSPAI 3.22 would not be important.

The results of one of the studies discussed in Section 3.3.10 provide more explicit evidence in this regard. Figure 35 compares the results using the TSPA model with the results for a model in which the effects of the unsaturated zone radionuclide transport barrier (and the saturated zone radionuclide transport barrier) are neglected. This scenario assumes that the radionuclides released from the engineered barrier system are discharged directly to the wells in the accessible environment. The results in Figure 35 show an increase in the estimate of mean annual dose;

however, the change is more than two orders of magnitude below the individual protection limit. Based on these results, the effects of both the unsaturated zone and saturated zone flow barriers, as they affect radionuclide transport, are not significant and therefore do not play a significant role in the postclosure performance.

More to the point of this particular KTI agreement, Figure 35 also shows the effect in which the unsaturated zone flow is included in the analysis, but additional aspects of the system affecting radionuclide transport (e.g., sorption, colloid filtration, matrix diffusion) are ignored. The results show very little difference between the results with the unsaturated zone and saturated zone radionuclide transport barriers fully neutralized and the results where the flow is included. The obvious conclusion is that the representation of the unsaturated zone flow system has a negligible effect in determining the transport of radionuclides. Accordingly, the information that would be provided in accordance with agreement TSPAI 3.22 is of little importance in determining whether the individual and groundwater protection performance objectives would be met.