RISK INSIGHTS BASELINE REPORT

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DIVISION OF HIGH LEVEL WASTE REPOSITORY SAFETY
OFFICE OF NUCLEAR MATERIAL SAFETY AND SAFEGUARDS
U.S. NUCLEAR REGULATORY COMMISSION

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ACRONYMS AND ABBREVIATIONS

°C	degrees Celsius
ACNW	Advisory Committee on Nuclear Waste
Am-241	americium-241
Am-243	americium-243
Bq	becquerel
BSC	Bechtel SAIC Company, LLC
CFR	Code of Federal Regulations
CHnv	Calico Hills non-welded vitric unit
Ci	curie
CNWRA	Center for Nuclear Waste Regulatory Analyses
DOE	U.S. Department of Energy
DRH	deliquescence relative humidity
EBS	engineered barrier system
ESF	exploratory shaft facility
FEPs	features, events and processes
FR	Federal Register
ft	foot
HLW	high-level radioactive waste
I-129	iodine-129
in	inch
ISI	integrated subissue
Kd	distribution coefficient
km	kilometer
KTI	key technical issue
m	meter
M	molar
mi	mile
mm	millimeter
MPa	megapascal
mrem	millirem
mSv	millisievert
NFE	near-field environment
Np-237	neptunium-237
NRC	Nuclear Regulatory Commission
PCSA	preclosure safety analysis
PRA	probabilistic risk assessment
Pu-239	plutonium-239
Pu-240	plutonium-240
RMEI	reasonably maximally exposed individual
SRM	staff requirements memorandum
SZ	saturated zone
Tc-99	technetium-99
TEDE	total effective dose equivalent
TPA	total-system performance assessment (NRC's computer code)
TSPA	total system performance assessment (DOE's computer code)
TSPA-LA	total system performance assessment for license application
TSPA-SR	total system performance assessment for site recommendation
TSPA-SSPA	total system performance assessment for the supplemental science and performance analyses

TSw	Topopah Spring welded tuff
UO_2	uranium dioxide
UZ	unsaturated zone

EXECUTIVE SUMMARY

As part of an ongoing effort to increase the use of risk information in its regulatory activities, the U.S. Nuclear Regulatory Commission (NRC) high-level radioactive waste (HLW) program is enhancing documentation of risk information and synthesizing the information to better support a risk-informed regulatory program. This effort is referred to as the Risk Insights Initiative. This report documents the results of the Risk Insights Initiative and provides the results in the form of the Risk Insights Baseline. The Risk Insights Baseline serves as a common reference for the staff to use in risk-informing the Agency's HLW program, as it continues through pre-licensing regulatory activities and prepares to review a license application that may be submitted by the U.S. Department of Energy (DOE) for a HLW repository at Yucca Mountain, Nevada.

The risk insights presented in this report address the staff's current understanding of the repository system following cessation of repository operations and permanent closure of the repository through the 10,000-year compliance period (i.e., the postclosure period). The risk insights are drawn from the staff's experience gained through the development and exercise of its total-system performance assessment (TPA) computer code; technical analyses conducted by the staff to support pre-licensing interactions with the DOE; and analyses conducted by the DOE and others. If the DOE submits a license application for a repository at Yucca Mountain, the staff will review the information provided by the DOE and make its determinations based on information available at that time.

Background on Risk-Informed Regulation

In the Probabilistic Risk Assessment (PRA) Policy Statement (60 FR 42622, August 16, 1995), the NRC formalized its commitment to risk-informed regulation through the expanded use of PRA in regulatory activities. In issuing the policy statement, the NRC expected the implementation to improve the regulatory process in three ways: (1) incorporation of PRA insights into regulatory decisions; (2) efficient and effective utilization of Agency resources; and (3) reduction of unnecessary burden on licensees. The PRA Policy Statement states, in part, "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." In a white paper staff requirements memorandum (SRM), "Risk-Informed and Performance-Based Regulation" (SRM for SECY-98-144; March 1, 1999), the NRC stated that a risk-informed approach should use risk insights to focus regulatory attention on issues, commensurate with their importance to public health and safety.

The definition for "risk" in the white paper takes the view that assessing risk involves three questions: "What can go wrong?" "How likely is it?" and "What are the consequences?" These three questions are referred to as the "risk triplet." The traditional definition of risk, that is, probability times consequences, is fully embraced by the "triplet" definition of risk. For the HLW postclosure repository system, the risk is usually expressed in terms of probability-weighted dose, for comparison to the dose-based individual protection standard.

In the HLW program, performance assessment is the risk-assessment methodology that is used to assess the risks associated with postclosure performance of a repository system. The HLW performance assessments include not only the system-level analyses performed to calculate probability-weighted dose, but also the supporting analyses performed to understand system-level results, the sensitivities and uncertainties in the system-level results, the capability of individual system components and processes, and interactions among the components and processes. Thus, risk insights may be explicitly expressed relative to system-level risk, such as probability-weighted dose and also may be expressed in terms of surrogate measures (e.g., calculation of waste package failure rates, release rates of radionuclides from the waste package, and transport times of radionuclides to the compliance location), as long as the relationship between the surrogate measure and the system-level risk is understood. In the HLW program, risk insights are based on the quantitative results of performance assessments.

Developing the Risk Insights Baseline

Building on approximately 20 years of performance assessment activity, the HLW Risk Insights Initiative began in 2002. The early efforts of the Risk Insights Initiative were aimed at enhancing communication, among the staff, of the more significant technical issues, and identifying and focusing the staff's attention on the more significant of the pre-licensing agreements that had been established between the NRC and the DOE. Following these initial efforts, the staff began the development of its risk insights baseline document. The risk insights baseline provides a system-level perspective on the relative significance of system features, events, and processes by looking at how they might affect the waste isolation capabilities of the repository system during the postclosure period, and the potential effect on public health and safety.

To facilitate the application of the risk insights baseline to its regulatory activities, the staff organized the risk insights around the 14 postclosure performance assessment model abstractions, referred to as integrated subissues (ISIs) (Figure ES-1). Two other primary NRC documents related to the HLW program, "Integrated Issue Resolution Status Report" (NUREG-1762) and the "Yucca Mountain Review Plan" (NUREG-1804), are also structured around these ISIs. Within each model abstraction, the staff developed individual risk insights to address the important features, events, and processes, both natural and engineered, of the repository system, and to communicate how they relate to waste isolation capability and to estimates of risk. The staff did not attempt to develop risk insights to address all the components of a potential repository system at Yucca Mountain, but has, instead, focused on those components the staff has identified as most significant to waste isolation.

The risk insights are generally framed around the three aspects of the risk triplet. Risk insights are generally stated in terms of a scenario, essentially a statement of the feature, event, or process that might exist or occur in the postclosure repository system. The baseline provides context for understanding the likelihood that the feature, event, or process will exist or occur during the compliance period. For each risk insight, the baseline provides: (1) supporting quantitative analyses, and an interpretation of the analyses to the extent necessary to explain the relationship between the analyses and the risk insight; and (2) a discussion of the uncertainties associated with the analyses and their interpretation.

To support the application of the risk insights baseline, the staff grouped the risk insights into three categories of relative significance (high, medium, and low), based on contribution to, or effect on, the waste isolation capabilities of the repository system. Three criteria were considered in evaluating the significance of the risk insights:

(1) Effect on the integrity of waste packages;
(2) Effect on the release of radionuclides from the waste form and waste package; and
(3) Effect on the transport of radionuclides through the geosphere and biosphere.

In general, high significance is associated with features, events, and processes that could: (1) affect a large number of waste packages; (2) significantly affect the release of radionuclides; or (3) significantly affect the transport of radionuclides through the geosphere or biosphere. Medium significance is associated with a lesser effect on waste packages, radionuclide releases, or radionuclide transport, and low significance is associated with no or negligible effect.

General Risk Insights on Geologic Disposal

Geologic disposal has been internationally adopted as an appropriate method for ensuring protection of public health and safety for very long time periods (e.g., 10,000 years) because deep geologic disposal: (1) limits the potential for humans to come into direct contact with the waste; (2) isolates the waste from a variety of natural, disruptive processes and events occurring on the surface of the earth; and (3) limits the transport of radionuclides, after release to ground water, by the natural hydrologic and chemical properties of geologic strata comprising a potential repository site. Additionally, it has been widely accepted that a geologic repository is to be comprised of "multiple barriers" as a means of providing defense-in-depth.

The inventory of HLW represents a significant risk, if the inventory were quickly released to the biosphere. However, current performance assessments of a potential repository at Yucca Mountain indicate that the majority (i.e., greater than 99 percent) of the inventory is isolated from man during the regulatory compliance period and beyond, because of the effectiveness of the engineered barriers and the attributes of the site (i.e., natural barriers). For example: (1) long-lived waste packages are expected to retain their integrity during the period of the highest thermal output of the waste, when the waste form behavior is most uncertain; (2) radionuclides are expected to be released slowly from the engineered barrier system once the waste packages degrade; and (3) radionuclides are expected to travel slowly from the engineered barrier system to the area where potential exposures might occur because of the sorptive properties of the surrounding rock. Thus, multiple barriers, as a defense-in-depth approach, result in a robust repository system that is more tolerant of failures and external challenges.

Table ES-1 provides a general perspective on the capabilities and effectiveness of the site and design attributes for isolating the radionuclides considered in the ground-water pathway. The site and design attributes are divided into three categories affecting waste isolation: (1) delay of the onset of initial release; (2) release rates from the engineered barriers (principally the waste package and waste form); and (3) transport in the geosphere. The effectiveness of each barrier is indicated by the letter "D" or "L," which is used to represent three levels of effectiveness by the number of letters present.

Table ES-1 offers a general explanation for the risk currently estimated for the proposed repository; namely, the variety and number of design and site attributes result in a very limited amount of the HLW inventory being transported by ground water to the compliance location. Additionally, from a defense-in-depth perspective, the importance of any one barrier is generally diminished as the number of relatively independent barriers increases. In other words, poor performance of one barrier does not cause a significant increase in the estimated risk; thus, confidence in the overall safety of the repository system is significantly enhanced when there are multiple and effective barriers.

Although the information presented in Table ES-1 provides a useful general overview of repository system capabilities for waste isolation, this approach does not address the uncertainties in estimating the behavior of the proposed repository system. The technical details and uncertainties are the subject of the detailed risk insights provided for the 14 ISIs (Figure ES-1). Additionally, Table ES-1 addresses releases in the ground-water pathway and does not address releases to the air pathway from a potential igneous event. Igneous activity has a potential for higher consequences than estimated for the ground-water pathway. However, the risk is still estimated to be small from this scenario because the probability for igneous activity is orders-of-magnitude below the probability for ground-water releases.

Detailed Risk Insights

The NRC staff have identified, to date, detailed risk insights related to performance of the repository system during the postclosure regulatory period. The risk insights are organized by the 14 performance assessment model abstractions (i.e., the ISIs) (Figure ES-1). For each risk insight, the NRC staff ranked the significance of the insight to waste isolation as means of providing a transparent view of the NRC's current understanding of features, events, and processes associated with a potential repository at Yucca Mountain. Such a representation of the risk insights benefits the NRC's HLW program by providing: (1) the NRC staff with information to risk-inform pre-licensing interactions and the review of a potential DOE license application; and (2) other stakeholders and interested groups and individuals (e.g., State of Nevada, DOE, Advisory Committee on Nuclear Waste) with information regarding the focus of the NRC's interactions with the DOE and its review of a potential license application. Table ES-2 summarizes the detailed risk insights, organized by the fourteen ISIs (Figure ES-1), along with their significance rankings.

Table ES-1. Representation of effectiveness of the attributes of waste isolation.

Radionuclide	Attributes of Waste Isolation						
	Onset of Release	Release Rate			Geosphere Transport		
	Waste Package	Waste Form	Solubility Limits	Solubility & Limited Water	Unsaturated Zone	Saturated Zone - Tuff	Saturated Zone - Alluvium
Americium-241	DDD				DDD	DDD	DDD
Plutonium-240	DDD			L	DDD	DD	DDD
Plutonium-239	DDD			L	DDD	DD	DDD
Americium-243	DDD			L	DDD	DD	DDD
Technetium-99	DDD	LL			D	D	D
Uranium-234	DDD			L	DDD	D	DDD
Nickel-59	DDD	LLL	L	LL	DDD	D	DDD
Carbon-14	DDD	LLL			D	D	D
Neptunium-237	DDD			L	DDD	D	DDD
Niobium-94	DDD	LL	LLL	LLL	D	DD	DDD
Cesium-135	DDD	LL			DDD	DDD	DDD
Selenium-79	DDD	LL			DD	D	DD
Uranium-238	DDD	L	LLL	LLL	DDD	D	DDD
Curium-246	DDD	L			D	DD	DDD
Iodine-129	DDD	LL			D	D	D
Thorium-230	DDD	LL	L	LL	DDD	DD	DDD
Chlorine-36	DDD	LL			DDD	DDD	DDD
Radium-226	DDD	LL		L	DDD	DD	DDD
Lead-210	DDD	LL	L	LL	DDD	DD	DDD

Notes: [D denotes delay time of at least 10,000 years (DDD); 1000 years (DD); and 100 years (D).]

[L denotes limit on release of 10,000 (LLL), 1000 (LL) and 100 (L) times less than 0.15 mSv (15 mrem).]

Table ES-2. Summary of Risk Insights Rankings: Significance to Waste Isolation.

ENG1 - Degradation of Engineered Barriers	
Persistence of a Passive Film	High Significance
Waste Package Failure Mode	Medium Significance
Drip Shield Integrity	Medium Significance
Stress Corrosion Cracking	Medium Significance
Juvenile Failures of the Waste Package	Low Significance
ENG2 - Mechanical Disruption of Engineered Barriers	
Effects of Accumulated Rockfall on Engineered Barriers	Medium Significance
Dynamic Effects of Rockfall on Engineered Barriers	Low Significance
Effects of Seismic Loading on Engineered Barriers	Medium Significance
Effects of Faulting on Engineered Barriers	Low Significance
ENG3 - Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms	
Chemistry of Seepage Water	High Significance
ENG4 - Radionuclide Release Rates and Solubility Limits	
Waste Form Degradation Rate	Medium Significance
Cladding Degradation	Medium Significance
Solubility limits	Medium Significance
Mode of Release from Waste Package	Low Significance
Effect of Colloids on Waste Package Releases	Medium Significance
Invert Flow and Transport	Low Significance
Criticality	Low Significance
UZ1 - Climate and Infiltration	
Present-day Net Infiltration Rate	Medium Significance
Long-term Climatic Change	Medium Significance
UZ2 - Flow Paths in the Unsaturated Zone	
Seepage	High Significance
Hydrologic Properties of the Unsaturated Zone	Medium Significance
Transient Percolation	Low Significance
UZ3 - Radionuclide Transport in the Unsaturated Zone	
Retardation in the Calico Hills non-welded vitric unit	Medium Significance
Matrix Diffusion in the Unsaturated Zone	Medium Significance
Effect of Colloids on Transport in the Unsaturated Zone	Medium Significance
SZ1 - Flow Paths in the Saturated Zone	
Saturated Alluvium Transport Distance	Medium Significance
SZ2 - Radionuclide Transport in the Saturated Zone	
Retardation in the Saturated Alluvium	High Significance
Matrix Diffusion in the Saturated Zone	Medium Significance
Effect of colloids on Transport in the Saturated Zone	Medium Significance
DIRECT1 - Volcanic Disruption of Waste Packages	
Probability of Igneous Activity	High Significance
Number of Waste Packages Affected by Eruption	High Significance
Number of Waste Packages Damaged by Intrusion	Medium Significance
DIRECT2 - Airborne Transport of Radionuclides	
Volume of Ash Produced by an Eruption	Medium Significance
Remobilization of Ash Deposits	Medium Significance
Inhalation of Resuspended Volcanic Ash	High Significance
Wind Vectors During an Eruption	Medium Significance
DOSE1 - Concentration of Radionuclides in Ground Water	
Well-pumping Model	Low Significance
DOSE2 - Redistribution of Radionuclides in Soil	
Redistribution of Radionuclides in Soil	Low Significance
DOSE3 - Biosphere Characteristics	
Characterization of the Biosphere	Low Significance

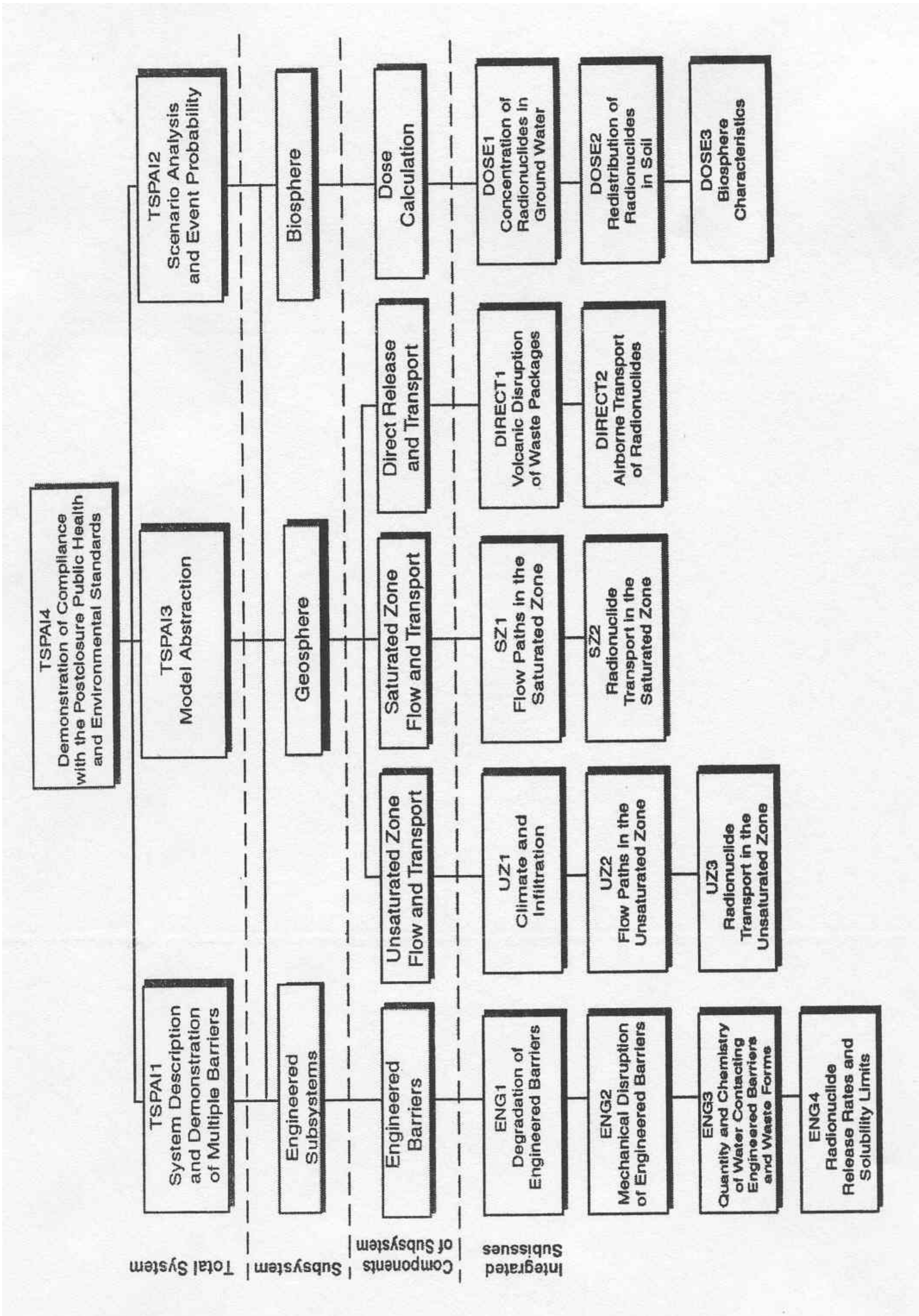

Figure ES-1. Components of performance assessment review. (From NRC. NUREG-1804, "Yucca Mountain Review Plan." Washington, DC: NRC. July 2003. Figure A1-5)

1. INTRODUCTION

As part of an ongoing effort to increase the use of risk information in its regulatory activities, the U.S. Nuclear Regulatory Commission (NRC) high-level radioactive waste (HLW) program is enhancing documentation of risk information and synthesizing the information to better support a risk-informed regulatory program. This effort is referred to as the Risk Insights Initiative. This report documents the results of the Risk Insights Initiative and provides the results in the form of the Risk Insights Baseline. The Risk Insights Baseline serves as a common reference for the staff to use in risk-informing the Agency's HLW program, as it continues through pre-licensing regulatory activities and prepares to review a license application that may be submitted by the U.S. Department of Energy (DOE) for a HLW repository at Yucca Mountain, Nevada.

The system description and the risk insights presented in this report are intended to assist the staff in its pre-licensing interactions with the DOE and in reviewing any license application the DOE may submit. The staff has not made any determinations regarding the technical conditions or the adequacy of a repository at Yucca Mountain at this time. If the DOE submits a license application for a repository at Yucca Mountain, the staff will review the information provided by the DOE, and make its determinations based on information available at that time.

1.1 Background

In the Probabilistic Risk Assessment (PRA) Policy Statement (60 FR 42622, August 16, 1995), the NRC formalized its commitment to risk-informed regulation through the expanded use of PRA in regulatory activities. In issuing the policy statement, the NRC expected the implementation to improve the regulatory process in three ways: (1) incorporation of PRA insights into regulatory decisions; (2) conservation of Agency resources; and (3) reduction of unnecessary burden on licensees. The PRA Policy Statement states, in part, "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy."

In its staff requirements memorandum for COMSECY-96-061 (April 15, 1997), the Commission envisioned a risk-graded approach where the staff would use risk information to focus on those licensee activities that pose the greatest risk to the public health and safety, thereby accomplishing the NRC's principal mission in an efficient and cost-effective manner. In general, activities of higher risk should be the primary focus of the Agency's efforts and resources.

The NRC defined the terms and expectations for risk-informed and performance-based regulation in a white paper staff requirements memorandum (SRM), "Risk-Informed and Performance-Based Regulation" (SRM for SECY-98-144; March 1, 1999). The guidance was intended to ensure consistent interpretation of the terms and implementation of the NRC's expectations with respect to their use in regulatory activities. The NRC believes that a risk-informed approach should use risk insights to focus regulatory attention on issues, commensurate with their importance to public health and safety. The NRC also stated in the white paper that risk insights can make the elements of defense-in-depth, a fundamental tenet of nuclear regulatory practice, more clear by quantifying them to the extent practicable.

Building on approximately 20 years of performance assessment activities, the HLW program staff began the Risk Insights Initiative, in 2002, to improve integration and communication of the issues considered important to the performance of the HLW repository, thereby enhancing its ability to conduct a focused, risk-informed review of both issue resolution activities and ultimately the license application that the DOE is anticipated to submit for the proposed geologic repository at Yucca Mountain.

In response to a request from the Commission, the staff provided an initial draft of the risk insights baseline to the Commission on June 5, 2003, along with an initial ranking of the risk significance of the key technical issue (KTI) agreements. This report provides the supporting quantitative analyses and discussion of uncertainties for the risk insights.

1.2 Purpose of the Risk Insights Baseline

The primary purpose of the risk insights baseline is to summarize the staff's current understanding of how a repository system at Yucca Mountain might function to isolate waste and, thus, protect public health and safety during the compliance period. The risk insights baseline outlines the staff's current thinking of how the principal features, events, and processes that might be present at Yucca Mountain following permanent closure of the repository could affect the estimated risks to an individual in the vicinity of Yucca Mountain following permanent closure. The staff perspective presented in the baseline is drawn from experience gained through its independent technical analyses, reviews, and performance assessment activities, as well as through the extensive technical interactions with the DOE and other groups external to the Agency, that have been completed to date. The risk insights baseline provides a common basis for the staff as it conducts its pre-licensing regulatory activities regarding postclosure repository performance in a risk-informed manner.

Risk insights are the results and findings drawn from risk assessments. In the HLW program, risk insights help to convey the significance of specific features, events, and processes to waste isolation capabilities and calculated estimates of system risk. The HLW risk insights have been integrated into a baseline and presented in a way that enhances their communication and understanding among the staff and others, both inside and outside the NRC. Although the significance of the risk insights may be expressed in somewhat qualitative terms (i.e., high, medium, and low significance), individual insights are supported by quantitative risk information derived from risk assessments and other technical analyses. The risk insights baseline summarizes these supporting analyses and discusses the associated uncertainties.

1.3 Objectives of the Risk Insights Baseline Report

The format of the risk insights baseline is intended to clearly communicate the staff's current understanding of the repository system, as supported by quantitative risk analyses. The baseline also discusses the relative uncertainties associated with the staff's understanding. Finally, the baseline organizes the risk insights into a structure that readily supports risk-informing the staff's HLW regulatory activities. To this end, the report:

- Clarifies the application of risk insights in the HLW program

- Documents the baseline set of risk insights for the postclosure repository system, identifying the significance of the insights relative to waste isolation

- Describes the quantitative analyses that support the risk insights

- Documents uncertainties associated with the baseline set of risk insights

- Discusses the application of the baseline set of risk insights to the staff's ongoing and future regulatory activities (e.g., issue resolution, license application review, etc.)

The individual risk insights were developed as concise statements to enhance their communication and comprehension. As stated, the staff developed the risk insights by drawing on many years of technical analysis and risk assessment experience.

1.4 Scope of the Risk Insights Baseline

The risk insights baseline presented in this report addresses the staff's current understanding of the repository system following cessation of repository operations and permanent closure of the repository through the 10,000-year compliance period (i.e., the postclosure period). This understanding is currently reflected, to a great extent, in the NRC's Total-system Performance Assessment (TPA) computer code. The staff, together with the Center for Nuclear Waste Regulatory Analyses (CNWRA), has developed this computer code in support of pre-licensing activities and potential review of a license application for a repository at Yucca Mountain. The risk insights baseline presented in this report is drawn from the staff's experience gained through the development and exercise of the TPA code, technical analyses conducted by the staff to support pre-licensing interactions with the DOE, and analyses conducted by the DOE and others.

The risk insights baseline presented in this report does not address the staff's current understanding of the risks associated with the operation of a repository before permanent closure (i.e., the preclosure period). The staff, together with the CNWRA, is currently developing the Preclosure Safety Analysis (PCSA) Tool, a computer code that the staff will use to support its review of preclosure safety issues. The risk insights baseline is expected to be expanded at a later date, to make use of the PCSA Tool results.

This report directly addresses risk insights related to estimating potential dose to an individual during the postclosure period. Although risk insights related to ground water protection are not explicitly identified in this report, the staff believes that they would be adequately addressed by the insights provided in this report.

2. APPLICATION OF RISK TERMINOLOGY IN THE NRC HLW PROGRAM

This chapter discusses the general application of risk terminology in the NRC's HLW regulatory program. The NRC's white paper report, "Risk-Informed and Performance-Based Regulation," is the basis for the risk terminology used in the HLW program. This section provides additional guidance on how the terms relate specifically to the risk insights baseline.

On March 1, 1999, the Commission approved issuing the "Risk-Informed and Performance-Based Regulation" white paper that outlines its expectations regarding the "risk-informed" and "performance-based" regulation of nuclear safety. The white paper defines these and other related terms in an effort to promote a more common understanding within the NRC and its regulated community, as well as by the public, as to how risk-informed and performance-based concepts apply to various Agency functions.

The Commission advocates using risk-informed and performance-based approaches in developing and implementing regulations, and directs the staff to increase its use of PRA in all regulatory matters to the extent supported by the state of the art in methods and data, and in a manner that complements the NRC's deterministic approach and supports the NRC's tradition of defense-in-depth. With respect to the HLW program, the NRC promulgated its regulation at 10 CFR Part 63 after the issuance of the PRA Policy Statement. Consistent with this policy, the regulation at 10 CFR Part 63 is risk-informed. The regulation includes risk-based requirements, as well as deterministic and prescriptive requirements. The staff is currently focused on risk-informed implementation of the rule.

This section discusses the application of several of the terms from the white paper to the HLW risk insights baseline. Much of the discussion has been taken directly from the white paper.

2.1 Risk

The definition for "risk" in the white paper takes the view that assessing risk involves three questions: "What can go wrong?" "How likely is it?" and "What are the consequences?" These three questions are referred to as the "risk triplet." The traditional definition of risk, that is, probability times consequences, is fully embraced by the "triplet" definition of risk.

For the HLW postclosure repository system, the risk is usually expressed in terms of probability-weighted dose, for comparison to the dose-based individual protection standard.

2.2 Risk Assessment

The white paper defines "risk assessment" as a systematic method for addressing the risk triplet as it relates to the performance of a particular system (which may include a human component) to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty. From this assessment, the important scenarios can be identified.

The method used to conduct a risk assessment depends on the particular system being evaluated. In the HLW program, performance assessment is used to assess the risks associated with postclosure performance of a repository system. Consistent with the white paper definition, the performance assessment methodology includes not only the system-level analyses performed to calculate probability-weighted dose, but also the supporting analyses performed to understand system-level results, the sensitivities and uncertainties in the system-level results, the capability of individual system components and processes, and interactions among the components and processes. These supporting analyses use intermediate results from system-level computer codes as well as results from auxiliary calculations.

2.3 Risk Insights

The white paper defines the term “risk insights” as the results and findings that come from risk assessments. Risk insights may be explicitly expressed relative to system-level risk, such as probability-weighted dose. Risk insights also may be expressed in terms of surrogate measures, as long as the relationship between the surrogate measure and the system-level risk is understood. The extent to which risk insights are explicitly factored into the activities and decision-making of a specific regulatory program depends on the maturity of risk assessment methodologies and data for that program. Incorporating risk insights into a regulatory program is intended to improve both efficiency and effectiveness of the regulatory program.

In the HLW program, risk insights are based on the quantitative results of performance assessments. As previously defined, performance assessments include the quantitative analyses of system-level performance (e.g., analyses using the TPA code or simplified models that result in a calculation of probability-weighted dose) as well as supporting analyses (e.g., calculation of waste package failure rates, release rates of radionuclides from the waste package, and transport times of radionuclides to the compliance location) that help the staff to understand the system-level results. Risk insights include the interpretation of and the conclusions drawn from the quantitative risk assessment results, relative to system-level risk (e.g., probability-weighted dose). Uncertainties in the risk estimates are addressed through the use of parameter ranges and alternative approaches and models.

The risk assessment methodologies applicable to the HLW program are relatively mature. Therefore, it is expected that HLW regulatory activities and decision-making will be guided by risk insights to a great extent, to improve effectiveness and efficiency.

2.4 Risk-Based and Risk-Informed Approaches

According to the white paper, a “risk-based” approach to regulatory decision-making is one in which such decision-making is solely based on the numerical results of a risk assessment. However, uncertainties in risk assessment methodologies and results limit the practicality and acceptability of purely risk-based regulatory decision-making. Because the Commission does not endorse risk-based regulatory decision-making, regulatory decision-making in the HLW program will not be based solely on the quantitative results of risk assessments.

The Commission endorses a “risk-informed” approach to regulatory decision-making, a philosophy whereby risk insights are considered, together with other factors, to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety. According to the white paper, a “risk-informed” approach enhances the deterministic approach, which requires safety systems capable of mitigating consequences of adverse conditions which are assumed to exist, by:

(1) Allowing explicit consideration of a broader set of potential challenges to safety;

(2) Providing a logical means for prioritizing these challenges, based on risk significance, operating experience, and/or engineering judgment;

(3) Facilitating consideration of a broader set of resources to defend against these challenges;

(4) Explicitly identifying and quantifying sources of uncertainty in the analysis (although such analyses do not necessarily reflect all important sources of uncertainty); and

(5) Leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions.

Where appropriate, a risk-informed regulatory approach can also be used to reduce unnecessary conservatism in purely deterministic approaches, or can be used to identify areas with insufficient conservatism in deterministic analyses and provide the bases for additional requirements or regulatory actions. "Risk-informed" approaches lie somewhere on the spectrum between the risk-based and purely deterministic approaches, depending on the regulatory issue under consideration.

Risk insights make the elements of defense-in-depth more clear by quantifying them to the extent practicable. Risk insights related to the individual performance of each defense system, in relation to overall performance, support decision-making on the adequacy of, or the necessity for, elements of defense. (See Section 2.5.)

The HLW program will follow a risk-informed approach to support regulatory activities and decision-making with respect to the requirements of the regulation at 10 CFR Part 63. This approach will focus resources on issues commensurate with their importance to risk. This approach will take into account the quantitative risk insights, together with uncertainties and sensitivities, engineering judgment, and other relevant factors.

2.5 Risk-Informed Approach and Defense-in-Depth

The concept of defense-in-depth has always been, and will continue to be, a fundamental tenet of regulating nuclear facilities. Defense-in-depth is an element of the NRC's safety philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. For the postclosure repository system, the regulation at 10 CFR Part 63 incorporates the defense-in-depth concept through the multiple barriers requirements. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges. Risk insights can make the elements of defense-in-depth more clear by quantifying their significance to waste isolation to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can assist and support regulatory decision-making. Decisions on the adequacy of, or the necessity for, elements of defense-in-depth, should reflect risk insights gained through identification of the individual performance of each safety system in relation to overall performance.

3. DEVELOPMENT OF THE RISK INSIGHTS BASELINE

3.1 Background

The Risk Insights Initiative began in January 2002. The early efforts of the Risk Insights Initiative were aimed at enhancing communication, among the staff, of the more significant technical issues, and identifying and focusing the staff's attention on the more significant of the pre-licensing agreements that had been established between the NRC and DOE. A facilitated approach was used during a series of meetings to develop a consensus, among the staff, of the significance of the 293 pre-licensing agreements relative to the calculated system-level risk estimates. During these initial activities, consideration of risk was not limited to a quantitative, dose-based definition. Instead, the staff considered more subjective measures of risk, as part of the enhanced communication effort.

The preliminary results of the initial Risk Insights Initiative exercise were presented to the Advisory Committee on Nuclear Waste (ACNW) in April 2002. The committee commented that the effort was successful as a communication exercise among the staff. However, the committee agreed with the staff's plan for repeating the exercise, but emphasized that the staff should focus on quantitative health and safety risks.

Following these initial efforts, the staff adopted a more integrated and quantitative systems approach to evaluating the significance of the KTI agreements. The staff began to develop the risk insights baseline, a concise description of how a repository system at Yucca Mountain might function during the postclosure period. The risk insights baseline provided a system-level perspective on the relative significance of system features, events, and processes, by looking at how they might affect the waste isolation capabilities of the repository system during the postclosure period, and the potential effect on public health and safety. The staff could then relate the agreements to this integrated system-level baseline to assess the relative significance of individual KTI agreements.

3.2 Development of the Current Baseline

The risk insights baseline has been developed by synthesizing available information drawn from quantitative risk assessments. The insights are based on many years of experience with conducting total system performance assessments, subsystems analyses, and auxiliary calculations. The insights are also based on the staff's review and interpretation of the performance assessments and supporting analyses conducted by others. Generally, the staff is relying on its own analyses to develop the risk insights, however, the analyses of others are useful for identifying alternative approaches and models that differ from those used by the staff in its performance assessment analyses. Understanding the analyses of others (e.g., DOE) is especially useful in specific areas where the staff's approach in its TPA computer code differ significantly from the approach of others (e.g., matrix diffusion in the unsaturated zone, representation of climate change). An understanding of the analyses of others provide additional information to evaluate the strengths and limitations of the risk insights.

The staff started with a system-level perspective and worked down to levels of greater detail and specificity. To coordinate the development of the set of insights, and to facilitate application of the risk insights baseline to the staff's regulatory activities, the staff organized the risk insights around the postclosure performance assessment model abstractions, referred to as integrated subissues (ISIs). These ISIs, are identified in two other primary NRC documents related to the HLW program, the "Integrated Issue Resolution Status Report" (NUREG-1762) and the "Yucca Mountain Review Plan" (NUREG-1804).

Within each model abstraction, the staff developed individual risk insights to address the important components (i.e., features, events, and processes, both natural and engineered) of the repository system and to communicate how these components relate to waste isolation capability and to estimates of risk. Thus, the risk insights discussed in this report for the HLW program tie the system components to some potential effect on health and safety, in terms of dose. The staff did not attempt to develop risk insights to address all the components of a potential repository system at Yucca Mountain, but has, instead, focused on those the staff has identified as most important.

The risk insights are generally framed around the three aspects of the risk triplet. Each risk insight is stated in terms of a scenario, essentially as statement of the feature, event, or process that might exist or occur. The baseline also provides context for understanding the likelihood that the scenario will exist or occur during the regulatory period of interest. The baseline also includes a discussion of the consequence of the scenario, in terms of its beneficial or adverse effect on the waste isolation capabilities of the repository system. An effect on waste isolation capability subsequently affects the estimated dose, or risk, to an individual.

Quantitative results for waste isolation capability (e.g., release rates from waste package, transport times for radionuclides) provide further insight to understand the effect on risk. Additionally, certain analyses were conducted beyond the regulatory period of 10,000 years to understand the sensitivity of repository performance to the timing of a process or event (i.e., when it occurs) that is currently estimated to occur beyond the compliance period (e.g., corrosion of the waste package, certain climate changes).

The baseline also provides an interpretation of the analyses to the extent necessary to explain the relationship between the analyses and the risk insight. These analyses generally have been excerpted from existing technical reports, papers, and presentations. The analyses address the likelihood of the condition described in the insight occurring at the site during the period of regulatory interest, or the consequence if the condition were to occur. Although development of the risk insights was based primarily on quantitative, system-level risk analyses, such as performance assessment calculations, the analyses include all of the supporting evidence that is used to build confidence in the calculations and the safety attributes of the repository system. Such evidence may include information from laboratory and field experiments, natural or man-made analogs, sensitivity analyses, and other specialized analyses at a subsystem level.

For each risk insight, the baseline also provides a discussion of the uncertainties associated with the analyses and their interpretation. Uncertainties are inherent in any attempt to characterize, understand, and model the future behavior of a natural or engineered system. Therefore, it is important that a discussion of the staff's current state of knowledge and understanding of the functioning of the repository system considers uncertainties. This includes data and model uncertainties, as well as uncertainties associated with the combined effects of scenarios. Generally, important uncertainties for estimating repository performance are addressed in the current analyses through a variety of approaches such as use of parameter ranges (e.g., range of retardation factors of radionuclides in alluvium) and conservative modeling approaches (e.g., assume southerly blowing wind direction for igneous activity). The discussion of uncertainties primarily provides insight where increased realism would reduce uncertainty in performance estimates. Because the approaches in the staff's TPA computer code tend to be conservative when the uncertainty is large, improved realism is generally expected to reduce the current estimate of risks.

The staff plans to use the risk insights baseline to help prioritize its pre-licensing activities, focus staff resources, and support risk-informed project management and decision-making in the HLW program, during pre-licensing activities and during the review of a potential license application for a Yucca Mountain repository. To support this intended application of the risk insights baseline, the staff grouped the risk insights into three categories of relative significance (high, medium, and low) based on contribution to, or effect on, the waste isolation capabilities of the repository system.

Although individual risk insights are supported by quantitative analyses, classifying the risk insights by relative significance to waste isolation is more qualitative. Staff judgment was used, as needed, when combining information from different analyses. Significance is evaluated relative to the waste isolation capabilities of the repository system. Three criteria were considered in evaluating the significance of the risk insights:

(1) Effect on the integrity of waste packages;
(2) Effect on the release of radionuclides from the waste form and waste package; and
(3) Effect on the transport of radionuclides through the geosphere and biosphere.

In general, high significance is associated with features, events, and processes that could: (1) affect a large number of waste packages; (2) significantly affect the release of radionuclides; or (3) significantly affect the transport of radionuclides through the geosphere or biosphere. Medium significance is associated with a lesser effect on waste packages, radionuclide releases, or radionuclide transport, and low significance is associated with no or negligible effect.

3.3 Multiple Barriers

Integral to developing the risk insights is the concept of multiple barriers (i.e., both engineered and natural barriers) in geologic disposal of HLW. For example, the safety of geologic disposal is enhanced if the system includes: (1) a long-lived waste package that retains its integrity during the period of the highest thermal output of the waste when the waste form behavior is most uncertain because of potentially high temperatures; (2) slow release rates of radionuclides from the engineered barrier system (EBS) once the waste packages are breached; and (3) slow travel of released radionuclides from the EBS to the area where potential exposures might occur. Multiple barriers, as an element of a defense-in-depth approach, results in a robust repository system that is more tolerant of failures and external challenges (e.g., poor or highly degraded performance is necessary in multiple areas to have a significant effect on risk). The risk insights are developed within the multiple barrier context (i.e., understanding the significance to waste isolation of the long-lived waste package, release rates of radionuclides, and transport of radionuclides in the context of the effect on risk estimates).

4. RISK INSIGHTS RELATED TO POSTCLOSURE PERFORMANCE ASSESSMENT MODEL ABSTRACTIONS

The system description and the risk insights presented in this report are intended to assist the staff in its pre-licensing interactions with the DOE and in reviewing any license application the DOE may submit. The staff has not made any determinations regarding the technical conditions or the adequacy of a repository at Yucca Mountain at this time. If the DOE submits a license application for a repository at Yucca Mountain, the staff will review the information provided by the DOE, and make its determinations based on information available at that time.

4.1 Risk Insights on Geological Disposal

Geologic disposal has been internationally adopted as an appropriate method for ensuring protection of public health and safety for very long time periods (e.g., 10,000 years) because deep geologic disposal: (1) limits the potential for humans to come into direct contact with the waste; (2) isolates the waste from a variety of natural, disruptive processes and events occurring on the surface of the earth; and (3) limits the transport of radionuclides, after release to ground water, by the natural hydrologic and chemical properties of geologic strata comprising a potential repository site. Additionally, it has been widely accepted that a geologic repository is to be comprised of “multiple barriers” as a means of providing defense-in-depth. The multiple barrier approach includes consideration of both natural barriers (e.g., hydrologic properties of rock and soil units, geochemical retardation) and engineered, or man-made, barriers (e.g., waste package, waste form) as a means to contain and isolate waste.

Understanding the potential risk of HLW begins by considering the radionuclides that comprise HLW, radionuclides that vary significantly with respect to inventory, radioactive half-life, and radiotoxicity. Table 4.1-1 provides information on the radionuclides relevant for evaluating releases in the ground-water pathway (i.e., radionuclides or daughters of radionuclides with radioactive half-lives of at least 100 years, and sufficient inventory, such that a portion of these radionuclides might be transported to the compliance location via ground water). As shown in Table 4.1-1, the overall radionuclide activity at 1000 years is dominated by relatively few radionuclides [i.e., americium-241 (Am-241), plutonium-240 (Pu-240), plutonium-239 (Pu-239), americium-243 (Am-243), and technetium-99 (Tc-99)]. The potential risk of the overall radionuclide inventory is determined by weighting the inventory of each radionuclide by its dose conversion factor, a measure of a radionuclide’s radiotoxicity. The potential risk of the inventory is similarly dominated by the same radionuclides, with the exception of Tc-99, which is not as significant because of its low radiotoxicity (i.e., low dose conversion factor).

Although the entire inventory of HLW represents a significant risk, if the inventory were quickly released to the biosphere, current performance assessments of a potential repository at Yucca Mountain indicate that the majority (i.e., greater than 99 percent) of the inventory is isolated from man during the regulatory period and beyond, because of the effectiveness of the engineered barriers and the attributes of the site (i.e., natural barriers). Evaluating the effectiveness of the multiple barriers requires an understanding of both the potential risk of the HLW inventory as well as the attributes of the design and site that affect the release and transport of each radionuclide. For example: (1) long-lived waste packages are expected to retain their integrity during the period of the highest thermal output of the waste, when the waste form behavior is most uncertain; (2) radionuclides are expected to be released slowly from the EBS once the waste packages are breached; and (3) radionuclides are expected to travel slowly from the EBS to the area where potential exposures might occur because of the sorptive properties of the surrounding rock. Thus, multiple barriers, as a defense-in-depth approach, result in a robust repository system that is more tolerant of failures and external challenges.

The risk insights for geologic disposal are developed by understanding the significance to waste isolation of the long-lived waste package, release rates of radionuclides, and transport of radionuclides, in the context of the effect on risk estimates. One approach for understanding and communicating the waste isolation capability of attributes of geologic disposal is to evaluate the releases of radionuclides at certain well-defined locations, such as from the waste package and geologic setting (the potential receptor location or compliance location). This helps characterize the behavior of specific barriers or subsystems of the overall repository. Figures 4.1-1 and 4.1-2 represent the “effective” activity released from the waste package and geologic setting, respectively. Effective activity is determined by weighting the activity for each radionuclide by its dose conversion factor, which allows the releases of radionuclides to be compared on a similar radiotoxicity basis. For the radionuclides shown in Figures 4.1-1 and 4.1-2, the majority of radionuclides that exit the waste package are not released from the geologic setting (i.e., at the compliance location) before 10,000 years. The radionuclides that tend to chemically sorb onto rock surfaces (e.g., plutonium, americium, neptunium) are not estimated to arrive at the compliance location, whereas, radionuclides such as iodine and technetium, which are less likely to chemically sorb, do arrive at the compliance location. The releases of iodine and technetium are barely discernable in Figure 4.1-2; however, the release rates from the geologic setting are approximately three times smaller than the release rates from the waste package. These figures provide quantitative risk information regarding the magnitude of releases from the waste package, and the attenuation of these releases, by the attributes of the geosphere, before they reach the compliance location.

Although the presentation of radionuclide release rates from specific subsystems is useful for understanding specific processes, this type of information does not readily convey the behavior of the spectrum of barriers of the repository system and the collective effectiveness for isolating waste. The staff has developed an approach for representing the waste isolation capabilities of specific attributes of the repository system in the context of the overall system, as a means to enhance understanding and risk insights. This approach represents the following three primary attributes for achieving waste isolation: (1) long-lived waste package; (2) slow release of radionuclides from the engineered barriers; and (3) slow migration of radionuclides in the geosphere. These attributes promote waste isolation by delaying and/or reducing releases of radionuclides to the compliance location. Performance assessment calculations are used to evaluate the effectiveness of individual barriers to isolate waste. For example, delay times are calculated for barriers that principally act to delay the onset of releases or the movement of radionuclides (e.g., waste package lifetime, transport time to move through the geosphere), whereas release rates are calculated for barriers that limit, rather than delay, releases (e.g., solubility limits, limited water contact with waste, spent fuel degradation rates).

Table 4.1-2 provides this type of general perspective on the capabilities of the site and design attributes for isolating the radionuclides considered in the ground-water pathway. The site and design attributes are divided into three categories affecting waste isolation: (1) delay for the onset of initial release; (2) release rates from the EBS (principally the waste package and waste form); and (3) transport in the geosphere. The effectiveness of each barrier associated with the attributes of waste isolation is indicated by the letter "D" or "L," which is used to represent three levels of effectiveness by the number of letters present.

When the design or site attribute delays the onset of release or transport in the geosphere, the level of effectiveness was determined according to delays of no less than 10,000 years (DDD); 1000 years (DD); or 100 years (D). For the release rate, where the attribute of a barrier is not a delay, but rather a limitation on the magnitude of the release, the level of effectiveness was determined by whether the magnitude of release, if instantly released to the biosphere, would result in a potential dose of 10,000 (LLL), 1000(LL), or 100 (L) times less than 0.15 millisieverts (mSv) [15 millirem (mrem)].

Table 4.1-2 offers a general explanation for the risk currently estimated for the proposed repository; namely, the variety and number of design and site attributes result in a very limited amount of the HLW inventory being transported by ground water to the compliance location. Additionally, from a defense-in-depth perspective, the importance of any one barrier is generally diminished as the number of relatively independent barriers increases. In other words, poor performance of one barrier does not cause a significant increase in the estimated risk; thus, confidence in the overall safety of the repository system is significantly enhanced when there are multiple and effective barriers.

The results presented in Table 4.1-2 are based primarily on the average behavior of the repository system and provide a useful general overview. However, this approach does not readily lend itself to addressing the uncertainties in estimating the behavior of the proposed repository system. For example, there are uncertainties with mechanical damage of the waste package, and the effect of colloidal transport of radionuclides, that are not directly represented in Table 4.1-2. The technical details and uncertainties are the subject of the detailed risk insights provided in the remainder of this section.

Additionally, Table 4.1-2 addresses releases in the ground water pathway and does not address releases to the air pathway from a potential igneous event. Igneous activity has a potential for higher consequences than estimated for the ground water pathway. However, the risk is still estimated to be small from this scenario, because the probability for igneous activity is orders-of-magnitude below the probability for ground water releases. (Sections 4.3.10 and 4.3.11 provide more discussion on igneous activity.)

Table 4.1-1. Inventory (based on the activity present at 1000 years) and weighted inventory (based on the activity present at 1000 years weighted by the dose conversion factor) of radionuclides evaluated in ground water releases.

Radionuclide	Half-Life, years	Inventory at 1000 years, % of total	Ground water Dose Conversion Factor, mrem/yr/pCi/l)	Weighted inventory at 1000 years, % of total
Americium-241	430	54	4.9	56
Plutonium-240	6,500	25	4.7	25
Plutonium-239	24,000	18	4.7	18
Americium-243	7,400	1.2	4.9	1.2
Technetium-99	210,000	0.73	0.0022	0.00033
Uranium-234	240,000	0.13	0.38	0.010
Nickel-59	76,000	0.12	0.00032	0.0000083
Carbon-14	5,700	0.065	0.0035	0.000048
Neptunium-237	2,100,000	0.064	6.0	0.080
Niobium-94	20,000	0.042	0.0096	0.000052
Cesium-135	2,300,000	0.027	0.012	0.000065
Selenium-79	65,000	0.023	0.013	0.000063
Uranium-238	4,500,000,000	0.016	0.35	0.0012
Curium-246	4,700	0.0032	4.9	0.0033
Iodine-129	16,000,000	0.0018	0.43	0.00016
Thorium-230	77,000	0.0011	0.74	0.00017
Chlorine-36	300,000	0.00058	0.0061	0.00000075
Radium-226	1,600	0.00019	1.8	0.000074
Lead-210	22	0.00019	7.3	0.00030

Table 4.1-2. Representation of effectiveness of the attributes of waste isolation.

Radionuclide	Attributes of Waste Isolation						
	Onset of Release	Release Rate			Geosphere Transport		
	Waste Package	Waste Form	Solubility Limits	Solubility & Limited Water	Unsaturated Zone	Saturated Zone - Tuff	Saturated Zone - Alluvium
Americium-241	DDD				DDD	DDD	DDD
Plutonium-240	DDD			L	DDD	DD	DDD
Plutonium-239	DDD			L	DDD	DD	DDD
Americium-243	DDD			L	DDD	DD	DDD
Technetium-99	DDD	LL			D	D	D
Uranium-234	DDD			L	DDD	D	DDD
Nickel-59)	DDD	LLL	L	LL	DDD	D	DDD
Carbon-14	DDD	LLL			D	D	D
Neptunium-237	DDD			L	DDD	D	DDD
Niobium-94	DDD	LL	LLL	LLL	D	DD	DDD
Cesium-135	DDD	LL			DDD	DDD	DDD
Selenium-79	DDD	LL			DD	D	DD
Uranium-238	DDD	L	LLL	LLL	DDD	D	DDD
Curium-246	DDD	L			D	DD	DDD
Iodine-129	DDD	LL			D	D	D
Thorium-230	DDD	LL	L	LL	DDD	DD	DDD
Chlorine-36	DDD	LL			DDD	DDD	DDD
Radium-226	DDD	LL		L	DDD	DD	DDD
Lead-210	DDD	LL	L	LL	DDD	DD	DDD

Notes: [D denotes delay time of at least 10,000 years (DDD); 1000 years (DD); and 100 years (D).]

[L denotes limit on release of 10,000 (LLL), 1000 (LL) and 100 (L) times less than 0.15 mSv (15 mrem).]

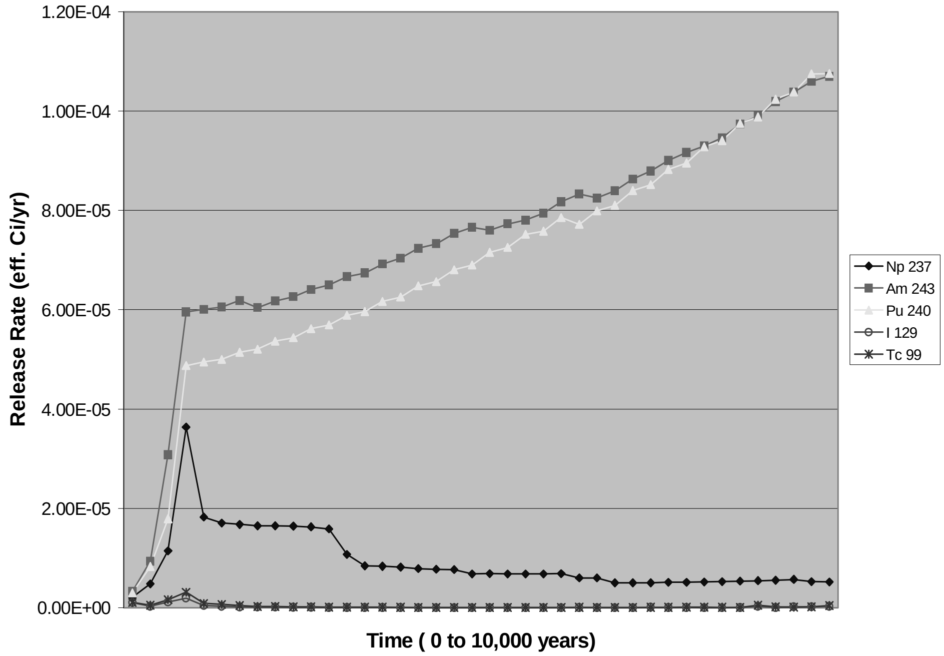

Figure 4.1-1. Effective activity released from the waste package.

Note: Effective activity determined by weighting release for each radionuclide by its dose conversion factor.

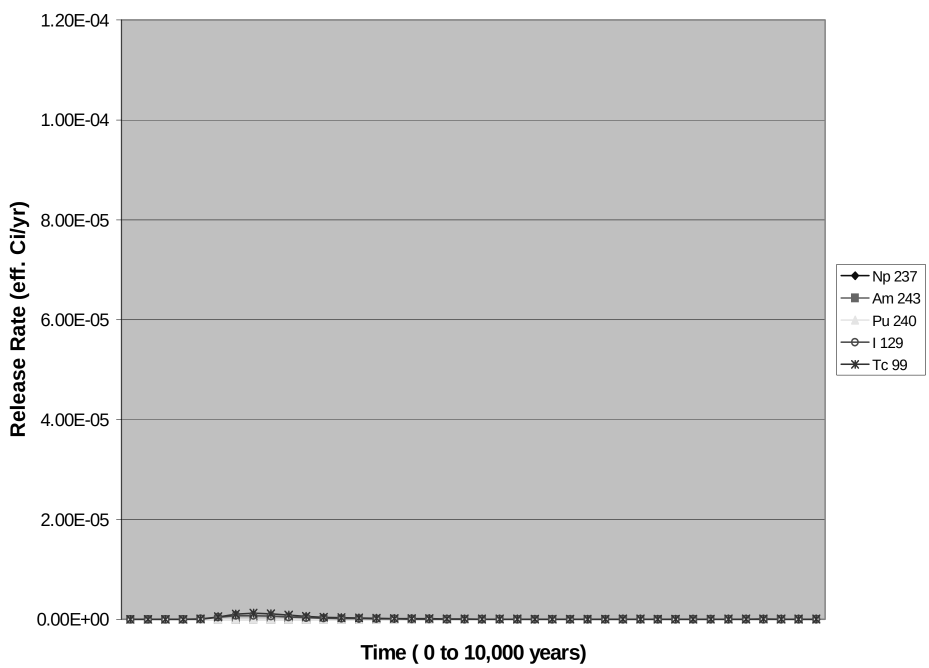

Figure 4.1-2. Effective activity released from the geologic setting.

Note: Effective activity determined by weighting release for each radionuclide by its dose conversion factor.

4.2 Current Understanding of the Postclosure Repository System

This section provides a summary of the staff's current understanding of a postclosure repository system at Yucca Mountain. This understanding is based on the process-level and system-level technical information and performance assessment results that are currently available. The staff's understanding continues to evolve as information becomes available through the pre-licensing activities and interactions.

The system description provided in this section is presented in seven sections:

- Infiltration, percolation, and seepage into the repository
- Degradation of the EBS, including the waste form
- Radionuclide release from the EBS
- Flow and transport of radionuclides in the unsaturated zone (UZ) below the repository
- Flow and transport of radionuclides in the saturated zone (SZ)
- Biosphere and reasonably maximally exposed individual (RMEI)
- Igneous activity

4.2.1 Infiltration, Percolation, and Seepage

Yucca Mountain has a semi-arid climate and currently receives an average of approximately 190 millimeters (mm) [7.5 inches (in)] of precipitation per year (yr). Future climate is expected to evolve according to anticipated glacial cycles. Evidence suggests at the last full glacial maximum, average, annual precipitation may have been 1.5 to 2.5 times larger than current climatic conditions, whereas average annual temperatures may have been 5 to 10 degrees Celsius ($^{\circ}$C) cooler. Approximately 95 percent of the precipitation currently falling onto Yucca Mountain is estimated to be removed by run-off, evaporation, and plant transpiration. The remainder infiltrates through the near-surface environment in a heterogeneous spatial pattern and generally percolates vertically downward through the unsaturated tuff toward the proposed repository horizon. However, large-scale features (e.g., fault zones, hydraulic conductivity contrasts at the interfaces between tuff layers) and small-scale features (e.g., variability of hydraulic conductivity within a tuff layer) may complicate the flow paths and cause deep percolation to redistribute spatially. Although surface infiltration is highly episodic, unsaturated flow reaching the repository is generally assumed to be steady and continuous as a result of the damping effect of the permeable and porous rock matrix of the overlying Paintbrush Tuff nonwelded horizons.

Deep percolation rates above the repository directly influences water seepage into the drifts and the amount of water entering breached waste packages, which, in turn, facilitates the release of radionuclides from the EBS into the UZ underlying the repository horizon. The ambient seepage model suggests that only a small fraction of the percolating water will enter the drift by way of dripping from the drift ceiling because of capillary diversion. The effect of the repository thermal pulse also affects seepage into drifts. When ventilation is stopped at the time of closure, the temperature of the wallrock will quickly rise. The quantity of percolating water that will reach the EBS will be significantly reduced as decaying radioactive waste heats surrounding rock above boiling temperature during the first few thousand years, thereby driving liquid water away. Water evaporated by repository heat will move toward cooler areas where it will condense and may flow back as liquid water toward the drifts (refluxing). A combination of ambient percolating and refluxed water may penetrate back along preferential flow paths, into the thermally perturbed rock, that has temperatures above boiling, and seep into the drifts. Not all areas of the drifts will experience above-boiling conditions. Edge-cooling effects allow water to be present in the wallrock throughout the performance period in drift areas near the periphery of the repository.

At early times following repository closure, the drifts will likely remain open and could be effective at diverting water (if water is present) around the drifts by capillary retention in the rock matrix and fracture networks. The drifts, however, may degrade over time and fill with rock-fall rubble from the tunnel walls. As a result, an increased fraction of the percolating water may contact engineered barrier components by way of seepage through rock rubble. Drift ceiling irregularities, such as small asperities and lithophysae, may give way to larger irregularities as drift degradation occurs, potentially reducing the amount of water diversion caused by capillary retention.

Film flow along open drift walls is another mechanism for diverting water away from the drip shield and waste package. The formation of a rubble pile in contact with waste packages or shields could provide an additional mechanism for percolating water to come into contact with engineered barriers. During the reflux period, or as the thermal pulse is dissipating, water seepage into the drifts (including along-wall seepage) contributes to the vapor pressure in the drifts and, thus, elevates the relative humidity of air surrounding the engineered components. Movement of water vapor into cooler areas of drifts could produce condensation that results in additional or focused dripping. Water dripping onto engineered components during the thermal period would evaporate and leave a residue, which, along with any dust present, may affect the chemistry of any liquid water, that is later present on those engineered components.

The chemistry of water contacting engineered components can strongly affect degradation of those components through aqueous corrosion processes. Estimating the evolution of the near-field environment is complex because of coupled thermal-hydrological-mechanical-chemical processes and changes in the emplacement drift configuration caused by the collapse and rubbling of overlying rocks. Water and gas compositions will be influenced by chemical reactions within the unsaturated fractured rock. Local changes in water and gas chemistry may result from interactions with engineered materials, corrosion products, or both. The presence of rubbled rock will cause higher temperatures within the near-field environment and engineered barrier components. Major processes affecting the evolution of the near-field environment include evaporative processes and mineral dissolution and precipitation, as well as aqueous- and gaseous-phase transport and chemical reactions.

Deep percolation rate also directly influences the transport of radionuclides through the UZ to the SZ. Transport of radionuclides through the UZ mainly occurs in fractures within the welded units and in the matrix within the nonwelded units. Sorption during matrix flow through vitric and devitrified nonwelded tuff horizons can significantly delay transport of radionuclides to the water table.

4.2.2 Degradation of the Engineered Barrier System

The current DOE design for the engineered components calls for 63,000 metric tons of commercial spent nuclear fuel, as well as 7000 metric tons of DOE spent nuclear fuel and solidified HLW to be loaded into waste packages before placement in tunnels cut into the unsaturated tuff approximately 350 meters (m) [1150 feet (ft)] below the surface. The commercial spent nuclear fuel generally is in the form of ceramic-like pellets of irradiated uranium-dioxide (UO_2) clad in corrosion-resistant Zircaloy tubes, approximately 0.6 to 0.9 mm (0.024 to 0.035 in) thick. The current waste package design for commercial spent nuclear fuel consists of a 20-mm (0.8-in) thick Alloy 22 outer container surrounding a 50-mm (2.0-in) thick type 316 nuclear-grade (NG) stainless steel inner container, to provide structural strength during preclosure operations. The staff's understanding is that after the spent nuclear fuel or other HLW is loaded, lids will be welded onto the waste packages before placing them in the repository; and before permanent closure of the repository, an inverted U-shaped metal drip shield, approximately 15-mm (0.6-in) thick, fabricated from titanium alloy grade 7, will be installed over the emplaced waste packages. The bulk of the 7000 metric tons of DOE waste will be in the form of borosilicate glass, poured into stainless steel canisters, and encased in waste packages of similar design to that used for commercial spent nuclear fuel.

The drip shield and waste package can protect the waste form from dripping water while they remain intact, thereby limiting both the timing and magnitude of radionuclide release. The drip shield may also limit the exposure of the waste package to aggressive chemical environments resulting from thermal-hydrological-chemical processes, as well as mitigate mechanical damage to the waste package from falling rocks. These engineered barriers may be compromised by various degradation processes, including corrosion and mechanical damage. The lifetime of the engineered barriers can be influenced by the environmental conditions to which they are exposed; rock-fall from drift degradation or seismicity; faulting; or ascending magma from volcanic activity.

The flow of water into a breached waste package will depend on the location and cross-sectional area of the breaches through the waste package. Four simplified categories of failure can facilitate understanding of the performance of the waste package in limiting radiological releases: (1) a limited number of waste packages with small cracks or perforations; (2) a small number of waste packages with large breaches; (3) a large number of waste packages with small cracks or perforations; and (4) a large number of waste packages with large breaches. A limited number of waste package breaches, either large or small, may result from aggressive and highly localized environments; isolated rock-fall from drift degradation or seismic events; faulting; and manufacturing defects. Stress corrosion cracking is the main process by which frequent but small cracking of waste packages could occur. The likelihood of stress corrosion cracking can be promoted by aggressive environmental conditions combined with residual stresses resulting from fabrication and closure operations or applied stresses as a consequence of extensive rock-fall from widespread drift degradation or seismicity, as well as accidental internal overpressure. Large widespread failures of the waste packages may result from accelerated, localized corrosion because of pervasive aggressive environments or extensive rock-fall from large-scale drift degradation or very large seismic events. In this context, the fabrication and closure processes may result in microstructural changes of the container material that can affect significantly the resistance to localized corrosion and mechanical damage, as well as the mode and extent of the resulting failure.

4.2.3 Radionuclide Release from the Engineered Barrier System

The EBS consists of the waste form; cladding (for spent fuel); pour container (for vitrified waste); waste package; invert; and drift. Once radionuclides released from the waste package leave the boundary of the drift, they are considered to be in the UZ, although components of the EBS, such as the invert, can also be considered to be unsaturated.

The waste form may begin to degrade once the waste package integrity is breached so that air, water vapor, and liquid water can come into contact with it. Although the waste form can degrade in the presence of air and water vapor, the release of most radionuclides of concern from the waste package and through the invert will only occur when liquid water is present to facilitate transport by means of advection (i.e., transport with the flow of water) and diffusion (i.e., transport from areas of high concentration to low concentration by random motion at the molecular level).

Spent nuclear fuel would be the main contributor to the radioactive inventory of the repository. This waste will be in the form of small UO_2 pellets placed in long tubes known as cladding. Most fuel will be clad in zirconium alloy (e.g., Zircaloy), which is highly corrosion-resistant, and will prevent the spent fuel from coming into contact with air and water as long as it remains intact. A small fraction of cladding will already be failed on placement in the waste packages. In addition, some reactor fuel was clad with aluminum or stainless steel, which has inferior or less-predictable corrosion resistance than zirconium. Factors affecting the long-term integrity of the Zircaloy cladding include: (1) localized corrosion and stress corrosion cracking of cladding that is exposed to in-package water; (2) hydride reorientation and cracking; (3) creep failure; and (4) mechanical stresses caused by seismic shaking, rockfall, faulting, and handling during shipment and loading. Under some conditions, small failures of the clad fuel-tubes will allow water and air to cause swelling of the exposed fuel pellets, which can lead to rapid “unzipping” of the cladding in that tube. Swelling of fuel pellets as well as hydride cracking and creep failure are processes that are likely to have a more pronounced effect on high burn-up fuel.

In the case of fuel reprocessing, vitrified glass-like waste will be encased in stainless-steel “pour canisters” before placement in the waste packages. The pour canisters, like cladding, will provide protection beyond what is available from the waste packages alone.

As the waste forms degrade and come into contact with liquid water, radionuclides will be released in the form of dissolved species, particulates, and colloids. The process of radionuclide release from the waste form to liquid water is generally known as “dissolution,” although it does not necessarily involve dissolved materials only. The water-borne radionuclides may then escape the waste package by advection and diffusion. Dissolution of radionuclides; formation of, or attachment to, mobile colloids; and incorporation into secondary minerals, formed from the degradation products of UO_2, may influence the transport rates and the amounts of radionuclides that are available for transport from the waste package. A small percentage of the waste consists of radionuclides that are very soluble and are expected to be released to the water quickly. The bulk of the radionuclide inventory consists of radionuclides that will be released from the waste form no faster than their solubility limit would allow. Therefore they will not be released from the waste package any faster than the flow of water at the solubility limit would allow. Some solubility-limited radionuclides (e.g., plutonium), particularly those associated with the vitrified HLW forms, can form colloids or attach themselves to naturally occurring or man-made, non-radioactive colloids (e.g., iron oxyhydroxides from corrosion of steel). Association of radionuclides with colloids can increase the effective concentration of the water above the solubility limit of the radionuclide itself. However, colloids behave differently from dissolved constituents, and may be inhibited from transport from the waste package by attaching to internal surfaces (Wilson, 1989). In addition, diffusion of colloids is significantly lower than truly dissolved materials because of their much greater size.

Once radionuclides exit the waste package, they must migrate from the waste package through the underlying invert to be released to the UZ. The current DOE design for the invert consists of a carbon steel support frame backfilled with compacted crushed tuff, up to about 0.5 m (20 in) in thickness, through which the radionuclides must migrate by advection and diffusion to the UZ. The porous nature of the invert material may delay transport of radionuclides; however, it is necessary to assess the porous flow and sorption properties of the invert material to determine the effectiveness of this barrier.

4.2.4 Flow and Transport of Radionuclides in the Unsaturated Zone below the Repository

The repository at Yucca Mountain will be underlain by approximately 300 m (1000 ft) of unsaturated volcanic rock layers above the water table. The series of unsaturated layers below the repository is comprised of tuffaceous rock exhibiting varying degrees of welding, which affect both the fracture density and matrix conductivity. Densely welded tuffs are brittle and typically develop interconnected fractures, which may allow water to divert around areas of lower conductivity, whereas non-welded tuffs exhibit low fracture density and higher matrix conductivity.

Dissolved and suspended radionuclides released from the engineered components would be transported by water flowing through the unsaturated tuffs to the water table. Water typically moves vertically downward through the unsaturated tuffs below the repository through a combination of fracture and matrix flow. However, large-scale (e.g., fault zones or hydraulic conductivity contrasts at the interfaces between tuff layers) and small-scale (e.g., the variability of hydraulic conductivity within a tuff layer) features may add complexity to the flow paths. Water tends to move slowly [e.g., currently estimated at 1 m (3.3 ft)/yr and slower] through unsaturated tuff layers when flow occurs predominantly within the rock matrix. As the water flux exceeds the matrix saturated hydraulic conductivity, water will flow through fractures. Water tends to flow more swiftly (e.g., an estimated tens of meters per year and faster) through tuff layers when flow is predominantly through fractures. Current understanding suggests the Calico Hills non-welded vitric (CHnv) layer is the only unsaturated tuff layer below the repository with sufficient matrix saturated hydraulic conductivity to allow water to flow predominantly within the rock matrix. The thickness of the CHnv layer is spatially uncertain and may “pinch-out,” resulting in no CHnv layer beneath portions of the repository.

In addition to the advective transport process described above, transport of radionuclides in the UZ would be affected by molecular diffusion between fractures and the rock matrix, mechanical dispersion, and physico-chemical processes such as sorption and precipitation. Sorption of radionuclides onto mineral surfaces can be a significant retardation mechanism when radionuclides move through the rock matrix because of the large surface area associated with the rock pores; conversely, fracture pathways have relatively limited surface area and thus exhibit limited if any sorption effects. Dissolved radionuclides transported by water within fractures may diffuse from the water within the fractures into the slow-moving water within the rock matrix, thereby limiting the transport of radionuclides in fractures. However, radionuclides transported by fracture flow could have limited time to diffuse from the fractures into the rock matrix because of the high velocity of water in the fractures (i.e., tens of meters per year).

Transport of radionuclides in colloidal form can limit the effectiveness of sorption processes; however, it can be expected that many colloids will be filtered out over long transport paths in geologic systems.

4.2.5 Flow and Transport of Radionuclides in the Saturated Zone

The SZ in the vicinity of Yucca Mountain consists of a series of alternating volcanic aquifers and confining units above the regional carbonate aquifer. The volcanic rocks generally thin toward the south and become interspersed with valley-fill aquifers to the south and southeast of Yucca Mountain. The valley-fill aquifer is composed of alluvium derived from Fortymile Wash, and colluvium from the adjacent highlands to the east and west, as well as lacustrine deposits formed near the southern end of Jackass Flats. The effective porosities of the fractured rock are expected to be lower than the valley-fill alluvium, resulting in higher ground water velocities in the fractured tuffaceous rocks.

Dissolved or suspended radionuclides released from the engineered components would be transported by water generally moving vertically downward through the unsaturated tuffs to the SZ. Ground water flow, in the SZ immediately below the repository, is driven by a small hydraulic gradient, approximately 0.0001, and is directed to the east-southeast through the eastward dipping upper volcanic confining unit and upper volcanic aquifer. Approximately 2 to 4 kilometers (km) [1.2 to 2.5 miles (mi)] east-southeast of Yucca Mountain, the hydraulic gradient is larger, approximately 0.001, and ground water is reoriented south through the tuff aquifer. South of Yucca Mountain, approximately 10 to 20 km (6 to 12 mi), radionuclides would enter the highly porous valley-fill aquifer.

The transport of radionuclides in the SZ would be affected by molecular diffusion between fractures and the rock matrix, mechanical dispersion, as well as physico-chemical processes associated with sorption of radionuclides onto mineral surfaces. Many radionuclides are retarded when moving through the porous alluvium, because of the large surface area associated with porous media. Certain radionuclides, however, such as iodine-129 (I-129) and Tc-99, are generally not retarded in geologic systems. Conversely, the fracture paths in the volcanic rock of the SZ are characterized by relatively limited surface areas and thus exhibit limited if any sorption effects within the fractures. However, dissolved radionuclides transported by water within fractures may diffuse from the water within the fractures into the slow-moving water within the rock matrix, thereby limiting the transport of radionuclides in fractures. The flow path in the SZ is more than 10 times longer than the flow path in the UZ (i.e., kilometers versus hundreds of meters). Therefore, significantly more time is available for radionuclides to diffuse from the fractures into the rock matrix of the SZ.

Transport of radionuclides in colloidal form or attached to colloids can limit the effectiveness of sorption processes; however, it can be expected that many colloids will be filtered out over long transport paths in geologic systems.

4.2.6 Biosphere and the Reasonably Maximally Exposed Individual

Radionuclides reaching the accessible environment enter the biosphere. The biosphere is the environment that the RMEI inhabits. Characteristics of the biosphere and the RMEI are based on current human behavior and environmental conditions in the Yucca Mountain region. Ground water transporting released radionuclides to the biosphere may be used for drinking and agricultural purposes typical of current Amargosa Valley practices. Radionuclides entering the biosphere via ground water may reach the RMEI through some combination of three likely pathways: direct exposure from surface or suspended contamination; inhalation of suspended dust that has been contaminated; or ingestion of contaminated water, plants, or animal products. In the igneous activity release scenario, radionuclides are assumed to transport through the air and over the land surface by remobilization processes to the RMEI location. Radionuclides entering the biosphere from an igneous event may reach the RMEI through pathways such as direct exposure from surface or suspended contamination; inhalation of suspended dust that has been contaminated; or ingestion of contaminated plants or animal products. Dose conversion factors from Federal guidance are used to convert exposures from contaminated materials to doses for the RMEI.

4.2.7 Igneous Activity

Basaltic igneous activity has occurred for over 10 million years throughout the Yucca Mountain region. The probability of future igneous activity occurring directly at the proposed repository site is presently estimated at between 10^{-7} and 10^{-8} per year; however, the discovery of additional buried anomalies thought to represent basalt could change this value. Igneous activity can affect the repository through direct release or indirect release of radionuclides during extrusive or intrusive events, respectively. Although the likelihood of future igneous activity is very small, the potential radiological doses are large enough to make a significant contribution to postclosure risk in current performance calculations.

If rising magma intersects repository drifts, the magma could flow into drift openings (intrusive event) and possibly continue an upward ascent to the surface (extrusive event). During the extrusive phase of an igneous event, magma reaches the surface and forms a volcanic eruption. Generally, a magma conduit to the surface gradually widens during an eruption. If this were to occur at Yucca Mountain, waste packages intersected by flowing magma in the conduit could be expected to break apart and allow the erupting magma to entrain radionuclides. These radionuclides would be transported downwind in the volcanic plume and deposited on the ground surface. Through time, wind and water could erode and redeposit this possibly contaminated ash. Potential radiological dose from extrusive igneous events results primarily from inhalation of contaminated ash.

During the intrusive phase of an igneous event, rising magma could flow into open or partially backfilled drifts in response to the pressure gradient between the confined magma and drift voids. The thermal, mechanical, and chemical environment in magma would likely damage the waste packages and may alter the HLW form. After the magma cools, radionuclides would then be available for potential release from damaged waste packages through the ground water pathway.

4.2.8 References

Wilson, C.N., and C.J. Bruton. "Studies on Spent-Fuel Dissolution Under Yucca Mountain Repository Conditions." UCRL-100223 Preprint. Indianapolis, Indiana: American Ceramic Society. 1989.

4.3 Baseline of Risk Insights

This section discusses the risk insights that have been identified to date by the NRC staff, related to performance of the repository system during the postclosure regulatory period.

The risk insights presented in this section are organized by 14 performance assessment model abstractions, also referred to as ISIs (Figure 4.3-1). This organizational format has also been used in two other primary NRC documents related to the HLW program, NUREG-1804 (NRC, 2003) and NUREG-1762 (NRC, 2002).

For each risk insight, this section provides a short title for the insight as well as a longer, more descriptive statement of the technical issue addressed by the insight. A ranking of the significance of the insight to waste isolation is provided. These rankings, which are based on risk insights, provide a transparent view of the NRC's current understanding of features, events, and processes associated with a potential repository at Yucca Mountain. Such a representation of the risk insights benefits the NRC's HLW program by providing: (1) the NRC staff with information to risk-inform its review of a potential DOE license application; and (2) other stakeholders (e.g., State of Nevada, DOE, ACNW) with information about the focus of the NRC's interactions with the DOE and review of a potential license application. For each risk insight, this section also provides a discussion of the technical basis for the insight, focusing as much as possible on supporting quantitative analyses as well as associated uncertainties.

Table 4.3-1 provides a summary of the risk insights, organized by the fourteen ISIs, along with their significance rankings. They are presented in the table in the order in which they are presented in the following sections.

4.3.0 References

NRC. NUREG-1762, "Integrated Issue Resolution Status Report." Washington, DC: U.S. Nuclear Regulatory Commission. July 2002.

NRC. NUREG-1804, "Yucca Mountain Review Plan." Revision 2. Washington, DC: U.S. Nuclear Regulatory Commission. July 2003.

Table 4.3-1. Summary of Risk Insights Rankings: Significance to Waste Isolation

ENG1 - Degradation of Engineered Barriers	
Persistence of a Passive Film	High Significance
Waste Package Failure Mode	Medium Significance
Drip Shield Integrity	Medium Significance
Stress Corrosion Cracking	Medium Significance
Juvenile Failures of the Waste Package	Low Significance
ENG2 - Mechanical Disruption of Engineered Barriers	
Effects of Accumulated Rockfall on Engineered Barriers	Medium Significance
Dynamic Effects of Rockfall on Engineered Barriers	Low Significance
Effects of Seismic Loading on Engineered Barriers	Medium Significance
Effects of Faulting on Engineered Barriers	Low Significance
ENG3 - Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms	
Chemistry of Seepage Water	High Significance
ENG4 - Radionuclide Release Rates and Solubility Limits	
Waste Form Degradation Rate	Medium Significance
Cladding Degradation	Medium Significance
Solubility limits	Medium Significance
Mode of Release from Waste Package	Low Significance
Effect of Colloids on Waste Package Releases	Medium Significance
Invert Flow and Transport	Low Significance
Criticality	Low Significance
UZ1 - Climate and Infiltration	
Present-day Net Infiltration Rate	Medium Significance
Long-term Climatic Change	Medium Significance
UZ2 - Flow Paths in the Unsaturated Zone	
Seepage	High Significance
Hydrologic Properties of the Unsaturated Zone	Medium Significance
Transient Percolation	Low Significance
UZ3 - Radionuclide Transport in the Unsaturated Zone	
Retardation in the Calico Hills Non-welded Vitric Unit	Medium Significance
Matrix Diffusion in the Unsaturated Zone	Medium Significance
Effect of Colloids on Transport in the Unsaturated Zone	Medium Significance
SZ1 - Flow Paths in the Saturated Zone	
Saturated Alluvium Transport Distance	Medium Significance
SZ2 - Radionuclide Transport in the Saturated Zone	
Retardation in the Saturated Alluvium	High Significance
Matrix Diffusion in the Saturated Zone	Medium Significance
Effect of Colloids on Transport in the Saturated Zone	Medium Significance
DIRECT1 - Volcanic Disruption of Waste Packages	
Probability of Igneous Activity	High Significance
Number of Waste Packages Affected by Eruption	High Significance
Number of Waste Packages Damaged by Intrusion	Medium Significance
DIRECT2 - Airborne Transport of Radionuclides	
Volume of Ash Produced by an Eruption	Medium Significance
Remobilization of Ash Deposits	Medium Significance
Inhalation of Resuspended Volcanic Ash	High Significance
Wind Vectors During an Eruption	Medium Significance
DOSE1 - Concentration of Radionuclides in Ground Water	
Well-pumping Model	Low Significance
DOSE2 - Redistribution of Radionuclides in Soil	
Redistribution of Radionuclides in Soil	Low Significance
DOSE3 - Biosphere Characteristics	
Characterization of the Biosphere	Low Significance

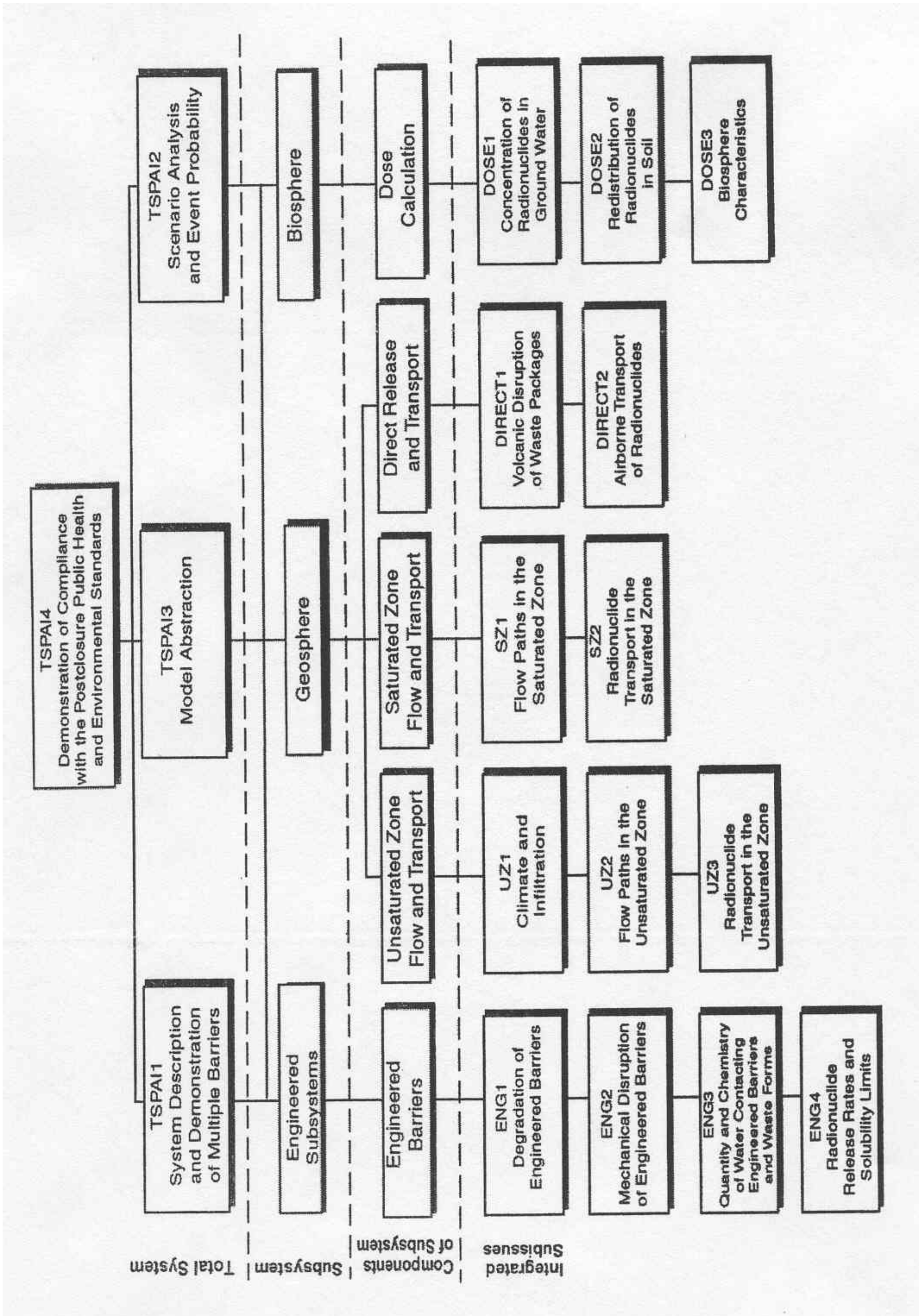

Figure 4.3-1. Components of performance assessment review. (From NRC, 2003, Figure A1-5)

4.3.1 Degradation of Engineered Barriers (ENG1)

Risk Insights:	
Persistence of a Passive Film	High Significance
Waste Package Failure Mode	Medium Significance
Drip Shield Integrity	Medium Significance
Stress Corrosion Cracking	Medium Significance
Juvenile Failures of the Waste Package	Low Significance

4.3.1.1 Discussion of the Risk Insights

Persistence of a Passive Film: High Significance to Waste Isolation

The persistence of a passive film on the surface of the waste package is anticipated to result in very low corrosion rates of the waste package. High temperatures and aggressive water chemistry conditions have a potentially detrimental effect on the stability of the passive film and may accelerate corrosion over extended surface areas.

Discussion

Under environmental conditions where a stable oxide film is maintained, corrosion is uniform and occurs at a slow rate. Typical values for the passive corrosion rate of titanium grade 7 and Alloy 22 are in the range of 10^{-5} to 10^{-3} mm/yr (10^{-7} to 10^{-5} in/yr) (Brossia, et al., 2001; Pensado, et al., 2002). Passive corrosion rates are generally independent of pH, redox potential, and solution composition, but exhibit an Arrhenius dependence on temperature (e.g., faster corrosion rates at higher temperatures).

Figure 4.3.1-1 shows the calculated dose assuming that: (1) in the base case, passive conditions prevail for all waste packages; and (2) passive conditions are not maintained for 25 percent of the waste packages. In this latter case, the expected dose approaches 0.01 mSv/yr (1 mrem/yr) after 10,000 years. As a result of the low corrosion rates under passive conditions (base case), the first waste package failure from uniform corrosion are estimated to occur after the 10,000 year-performance period. Doses that occur before 10,000 years in the base case estimate are the result of juvenile waste package failure. Assuming a uniform corrosion rate in the range from 5.0×10^{-5} to 5.4×10^{-4} mm/yr (2.0×10^{-6} to 2.1×10^{-5} in/yr), initial waste package failures from corrosion occur after 37,000 years and all waste packages fail after 403,000 years. The absence of passivity is assumed to result in uniform corrosion rates ranging from 5.0×10^{-3} to 5.4×10^{-2} mm/yr (2.0×10^{-4} to 2.1×10^{-3} in/yr). Failure times for waste packages in the absence of a protective, stable passive film are estimated to range from approximately 400 to 4000 years.

Uncertainties

The corrosion rate of the alloys proposed for the EBS, such as Alloy 22 and titanium alloys, are controlled by the presence of a thin, protective oxide film that restricts metal dissolution. The stability of oxide films on passive alloys is dependent on the material and exposure conditions. Loss of passivity can lead to corrosion rates that are orders-of-magnitude greater than those measured under passive conditions. For chromium-containing alloys such as Alloy 22, loss of passivity (depassivation) can occur in aggressive solutions characterized by low pH and high chloride concentrations, especially at high temperatures. See Section 4.3.3 for a discussion of the likelihood of such conditions. Chromium containing alloys exhibit transpassive dissolution at high anodic potentials, resulting also in high corrosion rates. Other processes, such as anodic segregation of sulfur or preferential dissolution of alloying elements, may also disrupt passivity. Passive films on titanium alloys are very stable in chloride solutions, but are strongly affected by the presence of fluoride, which significantly enhances the uniform corrosion rate (Brossia, et al., 2001). As discussed in Section 4.3.3, there are uncertainties associated with the amount and concentration of fluoride in the water seeping into the drift. Minor increases in fluoride concentration in pore water may have a significant effect on corrosion rate.

Waste Package Failure Mode: Medium Significance to Waste Isolation

The failure mode of the waste package (uniform corrosion, localized corrosion, or stress corrosion cracking) and its morphology (e.g., pits, cracks, or large corrosion holes or patches) is important for determining the amount of water that can enter the waste package.

Discussion

Different corrosion mechanisms create different types of failures (openings) in the surfaces of the engineered system. The amount of water that will enter a waste package and the amount of waste that will be transported out of a waste package will be influenced by the size of the openings. Intact surfaces will divert water away from the waste. The failure modes and morphology only have significant influence on the risk estimate when the openings are of limited surface area and frequency or when the water flow rates are very small. The degree of pessimism introduced into the analyses can reduce the effectiveness of geometrical considerations to limit the release of radionuclides. For diffusional releases, the size of the openings at which there is no longer a significant limitation on mass transport will be influenced by assumptions for boundary conditions and flow paths before radionuclides arriving at the opening and after they exit the opening. Current understanding indicates that stress corrosion cracks and pits that would form from localized corrosion will likely be small in cross-sectional area such that capillary forces may strongly limit the advective transport of water and radionuclides through such openings.

Alloy 22 was selected as the waste package outer container material to mitigate degradation and failure that can result from stress corrosion cracking, hydrogen embrittlement, localized corrosion, and accelerated uniform corrosion as a consequence of loss of passivity (CRWMS M&O, 2000a). Under conditions where the passive film can be maintained, failure of the waste package by corrosion-only processes is unlikely within the 10,000-year compliance period. Although highly resistant to various modes of corrosion, Alloy 22 is susceptible to localized corrosion, stress corrosion cracking, and hydrogen embrittlement.

Crevice corrosion of Alloy 22 can occur if aggressive solutions are in contact with the waste package and the corrosion potential exceeds the critical potential (i.e., the repassivation potential) for localized corrosion (Brossia, et al., 2001; Dunn, et al., 2003). In addition to the nature of the metal or alloy, the corrosion potential is mainly dependent on temperature and solution pH, whereas the repassivation potential for localized corrosion is dependent on temperature, the concentrations of aggressive and inhibiting species, and the microstructure of the material. Figure 4.3.1-2 shows the repassivation potential for Alloy 22 as a function of material condition and chloride concentration and the corrosion potential as a function of pH (Dunn, et al., 2003). In concentrated chloride solutions with a low pH, localized corrosion can be initiated when the corrosion potential exceeds the repassivation potential for localized corrosion. Localized corrosion is not expected in solutions with low chloride concentrations at neutral or alkaline pH, because the corrosion potential is below the repassivation potential. The localized corrosion susceptibility is also dependent on the relative concentrations of inhibiting and aggressive species. No localized corrosion will occur if a sufficient concentration of inhibiting species is present in the water contacting the waste package.

Figure 4.3.1-3 shows the expected dose for the base case and conditions where 10 percent of the waste packages are exposed to aggressive environmental conditions that promote localized corrosion. In the latter case, the peak expected dose is on the order of 0.001 mSv/yr (0.1 mrem/yr), during the first 10,000 years. The calculations shown in Figure 4.3.1-3 assume a limited amount of water can enter the waste package after corrosion penetrates the Alloy 22 outer container, which accounts for failure of the waste package from all corrosion processes.

Failure of the EBS components by uniform corrosion may result in large openings in the waste package. In contrast, pitting and crevice corrosion will likely result in small localized penetrations of the waste package. Stress corrosion cracks may also have a small aperture that will limit radionuclide release. The effects of localized penetrations of the waste package may be influenced by the combined effects of mechanical loads that are likely to occur as a consequence of rock fall, drift degradation, and seismic events. Mechanical loading of waste packages that are degraded as a consequence of uniform or localized corrosion, stress corrosion cracking, or hydrogen embrittlement may increase the size of the failure openings detrimentally affecting overall system performance.

Uncertainties

The amount of water that will enter a waste package and the amount of waste that will be transported out of a waste package will be influenced by the size of the openings. Release of radionuclides, in particular via diffusive mechanisms, will correlate with the failed surface area. Degradation modes that may lead to failure of the EBS and allow release of radionuclides include corrosion processes such as uniform and localized corrosion, stress corrosion cracking, and hydrogen embrittlement and mechanical interactions as a result of disruptive events. The location, size, and orientation of failure openings will be influenced by corrosion processes and mechanical interactions. However, there are uncertainties related to the specific characteristics of these openings, depending on the area of contact with water, the presence of deposits on the waste package surface, and the action and nature of the applied loads.

Drip Shield Integrity: Medium Significance to Waste Isolation

The integrity of the drip shield will influence the quantity and chemistry of the water that can develop on the waste package and the potential effects on corrosion modes and rates.

Discussion

The quantity and chemistry of the water contacting the waste package and the drip shield is addressed in Section 4.3.3 of this report. Analyses performed by the DOE for the total system performance assessment for site recommendation (TSPA-SR) (Figure 4.3.1-4) show the drip shield has little effect on repository performance (CRWMS M&O, 2000b). However, the role of the drip shield to control the formation of aggressive environments on the waste package surface was not included in the DOE model. Figure 4.3.1-5 shows the effect of drip shield integrity on the estimated dose, assuming 10 percent of the waste packages are exposed to environments that promote localized corrosion. Higher doses observed with accelerated drip shield failure are attributed to water seepage, which is not diverted by the drip shield, and hence contacts the breached waste packages.

Depending on the timing and extent of rockfall, the drip shield design may be important for limiting damage to the waste package (see Section 4.3.2.1).

Uncertainties

The drip shield is intended to limit ground water contact with the waste package. While the drip shield is intact, water that contacts the waste package may be limited to condensed water with low concentrations of aggressive species that are unlikely to promote localized corrosion or stress corrosion cracking. The drip shield will be constructed from titanium alloys that are resistant to localized corrosion in chloride solutions over a wide temperature range. However, the uniform corrosion rate of titanium alloys is dependent on the fluoride concentration. Faster corrosion rates and shorter failure times may occur on drip shield sections exposed to solutions with fluoride concentrations greater than 10^{-4} molar (M) (Brossia, et al., 2001; Lin, et al., 2003). However, titanium corrosion may be limited by the supply of fluoride from dripping water, and not strictly to the concentration threshold (Lin, et al., 2003). Failure of the drip shield by corrosion degradation mechanisms in combination with mechanical disruption may allow the formation of aggressive environments in contact with the waste package surface and lead to accelerated failure of the waste package. The formation of aggressive environments depends on many factors, with different degrees of uncertainties, that are related to the deposition of deliquescent salts, the rate of evaporation, and the composition of the seepage water. These aspects are discussed in Section 4.3.3.

Stress Corrosion Cracking: Medium Significance to Waste Isolation

Stress corrosion cracking of the drip shield or waste package affects a limited area and, thus, this corrosion process is not expected to allow substantial amounts of water to enter the waste package. However, applied loads arising from accidental internal overpressure, rockfall, or seismic events may increase the failure area and facilitate the ingress of water through the extended opening of stress corrosion cracks.

Discussion

The stress corrosion cracking susceptibility of Alloy 22 is dependent on material condition, corrosion potential, and stress intensity (Andresen, et al., 2003). Crack propagation rates for mill-annealed, cold-worked, and thermally aged Alloy 22 in basic saturated water are shown in Table 4.3.1-1. Under constant loading conditions, the crack propagation rates of the mill-annealed alloy decrease with time. However, sustained crack propagation under constant loading conditions has been observed for Alloy 22 in the cold-worked and thermally aged conditions. Recent results reported by General Electric and Lawrence Livermore National Laboratory and results obtained in independent tests conducted at the CNWRA show that

Alloy 22 is susceptible to stress corrosion cracking in simulated concentrated water at temperatures below boiling, provided sufficient stress intensity is present.

The effect of stress corrosion cracking on the estimated dose is likely to be low because of the limited area of the cracks. In Figure 4.3.1-6, mean values (from 100 realizations) of radionuclide release rates [I-129, Tc-99, and neptunium-237 (Np-237)] from the waste package are compared to estimates of diffusive release. Case 1 (continuous lines) is the reference case which assumes 90 percent of the waste packages are breached by general corrosion and 10 percent by localized corrosion. Case 2 (lines with circles) considers that radionuclides are released from the waste package only through stress corrosion cracks by a diffusive mechanism. In deriving these release rates (lines with circles), the following assumptions were made: (1) all the waste packages are breached by stress corrosion cracking at the time of emplacement; (2) a thin film of water connects the spent fuel with the exterior of the waste package; (3) the radionuclide concentration in the film, at the point of contact with the spent fuel, is determined by the saturation of radionuclide-bearing solids; (4) the concentration at the end of the diffusive path is zero; (5) the problem is one-dimensional with a path length equal to 0.3 m (1.0 ft) and film-cross section of 10^{-8} m^2 (10^{-7} ft^2); and (6) no credit is taken for cladding protection. As seen in Figure 4.3.1-6, diffusive releases, as modeled, are dominant during the first 4000 years; however, they are at least one order of magnitude below the maximum release rates of Case 1 (I-129, Tc-99), occurring at around 10,000 years. The estimated diffusive releases of Np-237 are less than 10^5 becquerels (Bq)/yr [10^{-5} curies (Ci)/yr] and are not displayed in Figure 4.3.1-6. After approximately 4000 years, the radionuclide release rates associated with Case 1 exceed the diffusive release rates.

Advective release through stress corrosion cracks could occur if cracks were opened by mechanical loading (e.g., mechanical interactions of the waste package with other components of the engineer barrier system during seismic events or as a result of static loading). Figure 4.3.1-7 shows the estimated effects of advective release of Np-237, I-129, and Tc-99 through cracks. To facilitate comparison, Figure 4.3.1-7 includes radionuclide release rates of Case 1 from Figure 4.3.1-6. Stress corrosion cracks were assumed to develop during a period of temperatures above the boiling point of water and relative humidity above a deliquescence point of salt formation. Radionuclide release rates from the waste package were estimated for the flow-through scenario, and decreased by a factor of 1/1000 to account for protection against seepage offered by the unbreached area of the waste package. Protection by the drip shield or cladding was disregarded. Under these assumptions, it was estimated that advective release through stress corrosion cracks was largest during early years of repository operation, up to 4000 years (Case 3, lines with circles in Figure 4.3.1-7). However, maximum release rates of Np-237, Tc-99, and I-129 are one order of magnitude lower than those derived for Case 1 (continuous lines).

Uncertainties

Stress corrosion cracking requires the combination of a susceptible material or microstructure, an aggressive environment, and an applied or residual tensile stress. Although nickel base alloys are known to be resistant to environmentally assisted cracking in hot chloride solutions, stress corrosion cracking of Alloy 22 has been reported in simulated ground water solutions that may contact the waste packages (Andresen, et al., 2003; King, et al., 2002; Estill, et al., 2002).

Stress corrosion cracks that penetrate the waste package may be tight and may restrict the transport of water into the waste package. Cracks that remain tight may allow only diffusive release of radionuclides. Stress corrosion cracking coupled with mechanical loading from disruptive events may propagate existing cracks or enlarge existing failures and allow advective release of radionuclides.

There is significant uncertainty, associated with crack geometry and the effect of applied loads, arising from accidental internal overpressure, rockfall, or seismic events that may propagate existing stress corrosion cracks, or lead to mechanical failure of the degraded waste package.

Release rates associated with stress corrosion cracking may be influenced by: (1) limited cross-section of stress corrosion cracks; (2) frequency of nucleation sites for stress corrosion cracks; (3) frequency of chemical solutions necessary for stress corrosion cracking; (4) levels of material stress needed for stress corrosion cracking; (5) stress corrosion cracking propagation rate; and (6) crack location on the waste package geometry.

Juvenile Failures of the Waste Package: Low Significance to Waste Isolation

Juvenile or early failures of the waste package (e.g., closure welding defects, such as flaws, which can promote other degradation processes) are expected to be limited to a small fraction of waste packages and not have a significant effect on waste package performance and hence on radionuclide release.

Discussion

The base case results from the TPA code, and the results from the DOE's total system performance assessment for the supplemental science and performance analyses (TSPA-SSPA) model, quantitatively support that juvenile or early failures of the waste package are of low significance to waste isolation (Figure 4.3.1-8 and Figure 4.3.1-9). For the TPA Version 4.1 base case, the risk from the nominal scenario (on average 44 juvenile failures, or 0.63 percent) is 0.00021 mSv/yr (0.021 mrem/yr) at 10,000 years. The DOE's TSPA-SSPA model had on average less than one package failure per stochastic realization, and the resultant doses were very small at 10,000 years [e.g., less than 10^{-6} mSv/yr (10^{-4} mrem/yr)]. Quality assurance procedures for fabrication, characterization, handling, and emplacement of waste packages should reduce the likelihood of significant defects and, therefore, juvenile failures. Quantitative analyses to date demonstrate the repository system is likely to tolerate limited waste package failures. Processes such as loss of passivity or occurrence of localized corrosion are not considered to be part of juvenile failures and are modeled separately.

Uncertainties

Initial defects coupled with waste package degradation processes and mechanical loading as a result of disruptive events may lead to early failures of waste packages. The number of waste packages that are susceptible to early failure processes will be dependent on the frequency, type, size, and orientation of the initial defects.

4.3.1.2 References

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Table 4.3.1-1. Crack Propagation Rates for Alloy 22 in Basic Saturated Water

Material Condition	Stress Intensity, $MPa \cdot m^{1/2}$	Crack propagation Rate, mm/s
Mill-annealed	30	0
	45	4×10^{-10} — 1.3×10^{-9}
Mill-annealed plus 20-% cold work	30	2×10^{-10} — 5×10^{-10}
Thermally aged 175 hours at 700 °C (1292°F)	16.5	—
	24.2	8×10^{-10} — 1.3×10^{-9}

(From Andresen, et al., 2003)

NOTES: English equivalents for $MPa \cdot m^{1/2}$ and mm/s are as follows: $MPa \cdot m^{1/2} \times 0.9091$ = ksi $in^{1/2}$ and mm/s × 0.039 = in/s.

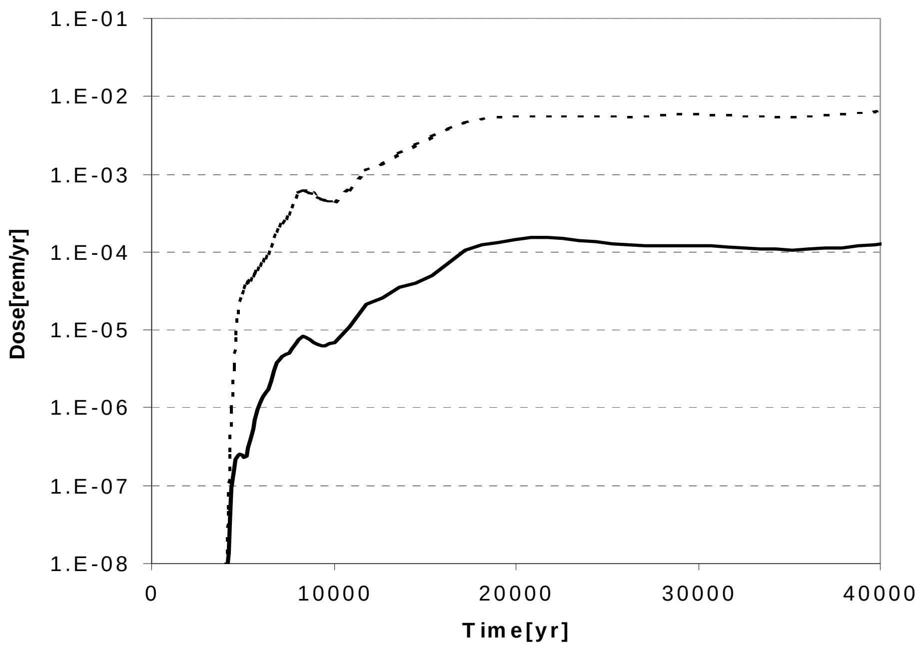

Figure 4.3.1-1. Dose calculations as a function of waste package corrosion rate. In the base case (TPA 4.1j) (continuous line) all waste packages are assumed to undergo passive dissolution. Assuming enhanced general corrosion rate for 25 percent of the waste packages (broken line) results in an increase in the calculated dose.

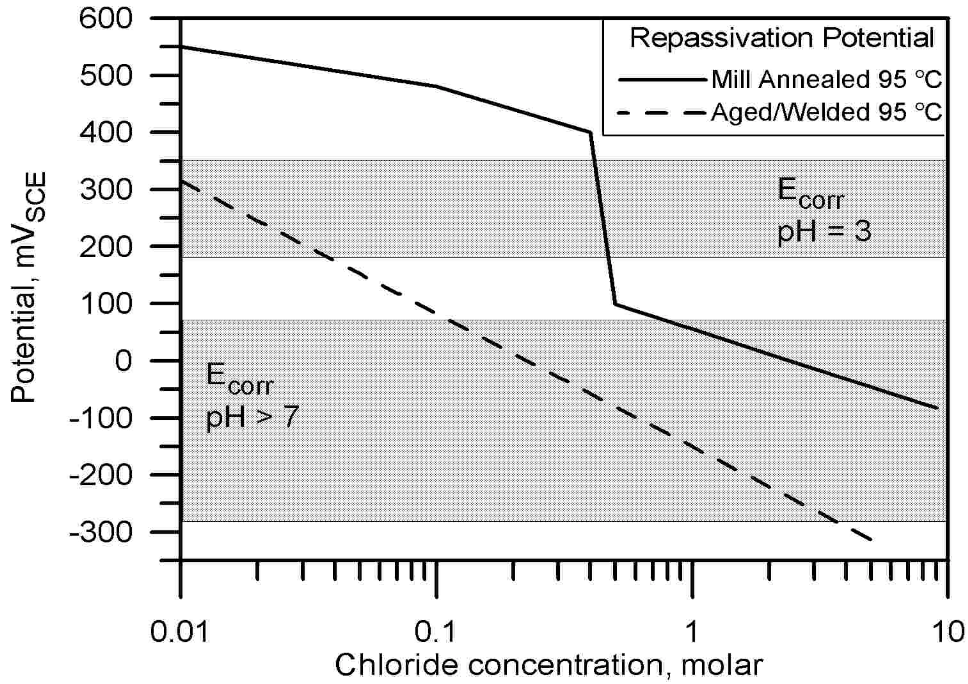

Figure 4.3.1-2. Repassivation potential for crevice corrosion for mill annealed and thermally aged or welded Alloy 22 as a function of chloride concentration at 95°C. Range of corrosion potentials as a function of pH are shown as shaded bands.
(From Dunn, et al., 2003)

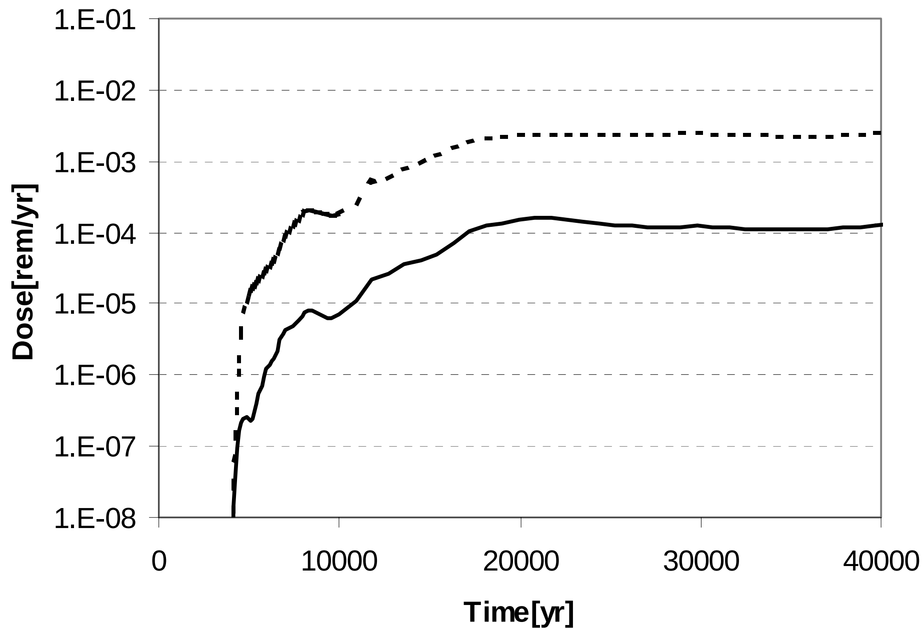

Figure 4.3.1-3. Effect of waste package corrosion mode on calculated dose. All waste packages are assumed to undergo passive dissolution in the base case (continuous line). Assuming 10 percent of the waste packages are exposed to aggressive solutions that promote localized corrosion increases the calculated dose (broken line).

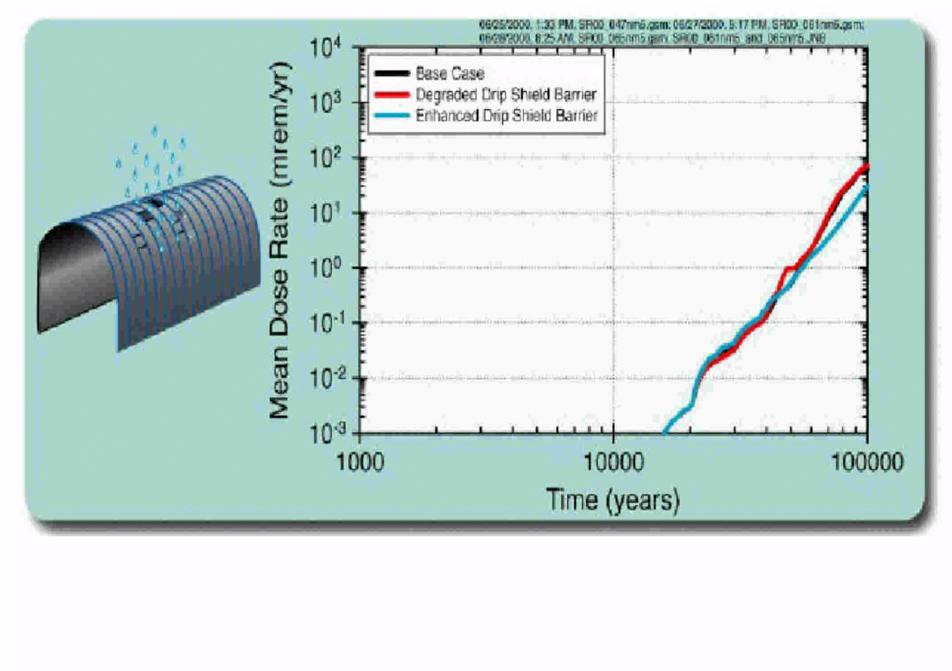

Figure 4.3.1-4. Drip shield sensitivity analysis using the total system performance assessment for site recommendation (TSPA-SR). (From CRWMS M&O, 2000b, Figure 5.3-3)

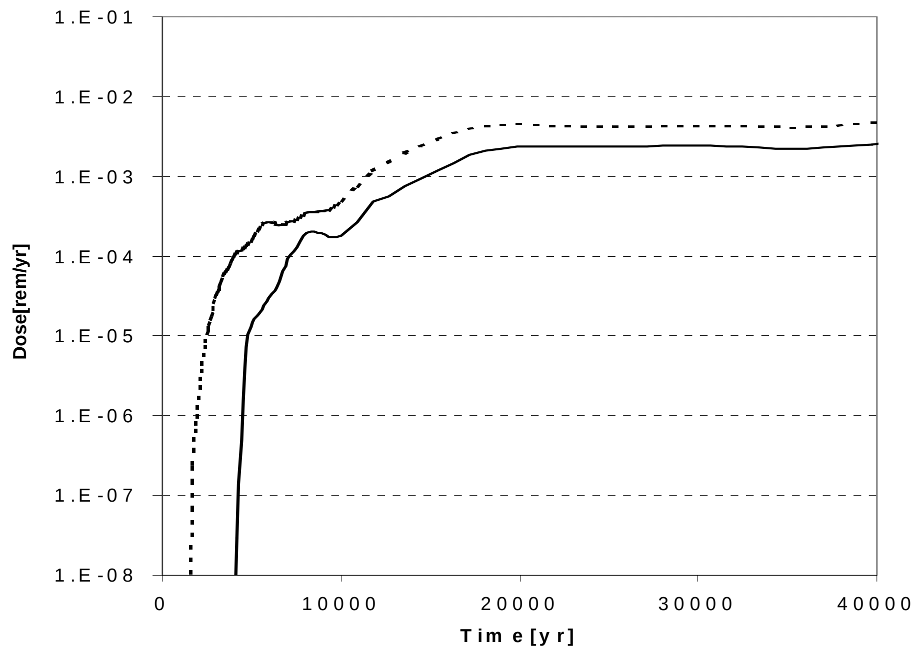

Figure 4.3.1-5. Effect of drip shield failure on calculated dose. In the reference case (10 percent of waste packages are assumed to display localized corrosion) drip shield failures are assumed to have a log normal distribution with 0.1 percent of drip shields failing at 2700 years and 99.9 percent failing at 20,400 years (continuous line). Enhanced failures of drip shields assume a uniform distribution with 10 percent of the drip shields failing between 1000 and 5000 years (broken line).

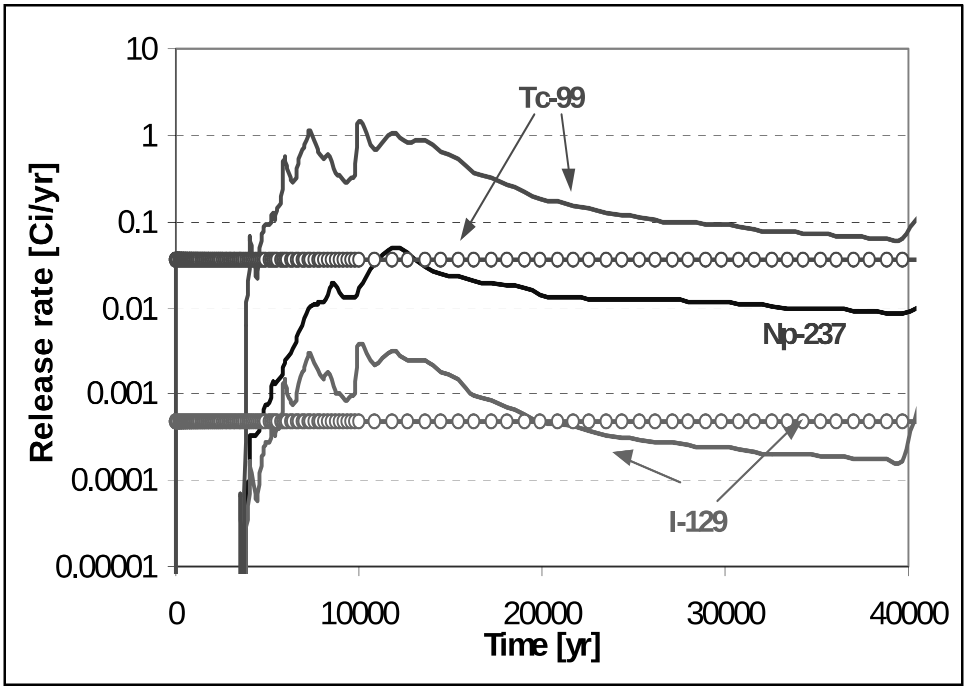

Figure 4.3.1-6. Radionuclide release rates from the waste package for two cases. Case 1 (continuous lines) is a reference case accounting for 90 percent of the waste packages breached by general corrosion and 10 percent by localized corrosion. Case 2 (lines with circles) shows diffusive release rates (I-129 and Tc-99) assuming saturation at the end of the water film in contact with the spent fuel, and zero concentration at the other end. [technetium-99 (Tc-99); iodine-129 (I-129); neptunium-237 (Np-237)]

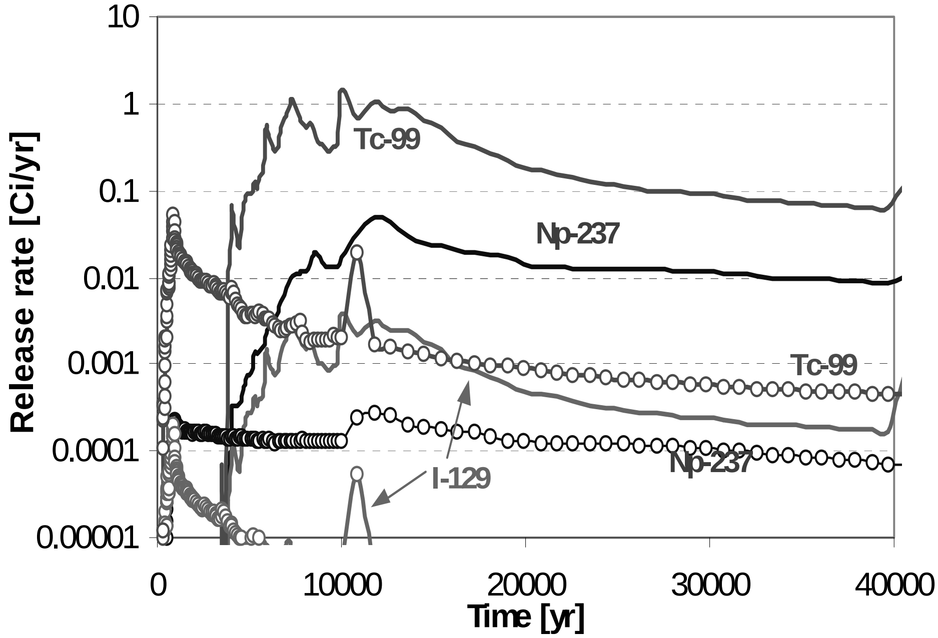

Figure 4.3.1-7. Radionuclide release rates from the waste package for Cases 1 and 3. Case 1 (continuous lines) is a reference case accounting for 90 percent of the waste packages breached by general corrosion and 10 percent by localized corrosion. Case 3 (lines with circles) shows advective release rates estimated by multiplying the waste package release rates in the flow-through scenario by a factor of 1/1000 (factor to account for protection against seepage offered by unbreached waste package surface). Both curves correspond to mean values from 100 realizations. [technetium-99 (Tc-99); iodine-129 (I-129); neptunium-237 (Np-237)]

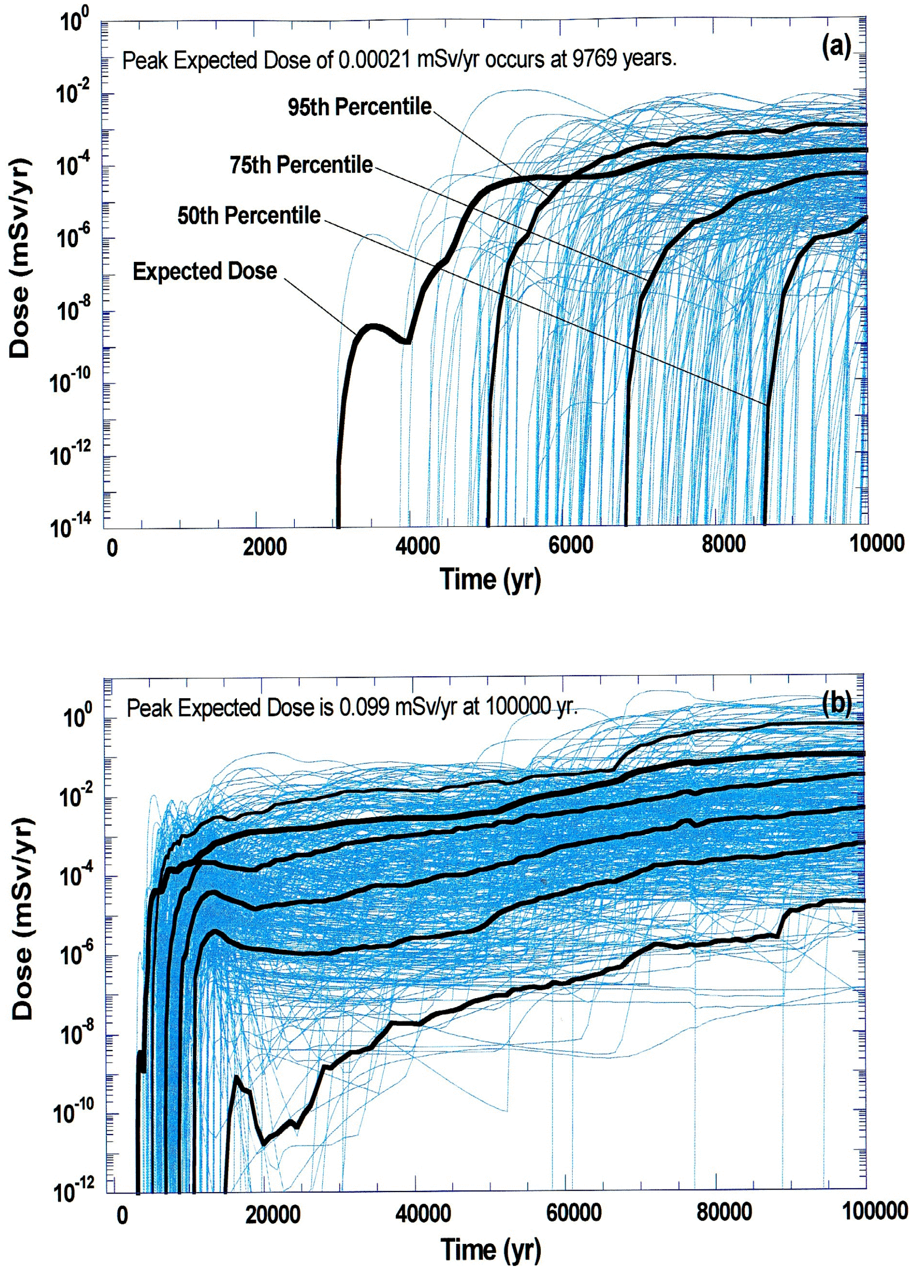

Figure 4.3.1-8. Ground water dose in: (a) 10,000 years; and (b) 100,000 years, including the average dose for 350 realizations. (From Mohanty, et al., 2004, Figure 3-20)

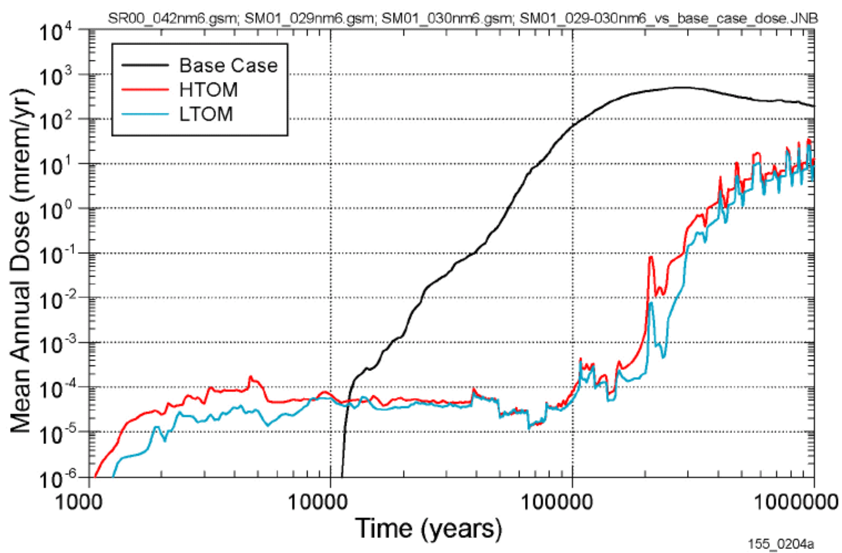

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Figure 4.3.1-9. Summary of peak dose performance results. [HTOM: High Temperature Operating Mode; LTOM: Low Temperature Operating Mode; Base Case: TSPA-SR] (From BSC, 2001, Figure 4.1-1)

4.3.2 Mechanical Disruption of Engineered Barriers (ENG2)

Risk Insights:	
Effects of Accumulated Rockfall on Engineered Barriers	Medium Significance
Dynamic Effects of Rockfall on Engineered Barriers	Low Significance
Effects of Seismic Loading on Engineered Barriers	Medium Significance
Effects of Faulting on Engineered Barriers	Low Significance

4.3.2.1 Discussion of the Risk Insights

Effects of Accumulated Rockfall on Engineered Barriers: Medium Significance to Waste Isolation

Mechanical loading from rockfall rubble accumulated from drift degradation over time may lead to failure of the drip shields and waste packages. The failure of the drip shields and waste packages will depend on the rate of accumulation of rockfall rubble in the drift (building static load on the drip shield) and the threshold load-bearing capacity of the drip shields and the waste packages. The accumulation of rock rubble in the drift outside the drip shield will also increase the waste package and drip shield temperatures.

<u>Discussion</u>

Current understanding of the degradation of mined openings indicates that all of the drip shields could experience static loads from rockfall rubble that can accumulate after repository closure. These loads are expected to damage the drip shields. A TPA calculation (Figure 4.3.2-1) indicates that if all drip shields fail from rockfall without any associated waste package failure, the dose consequence will be low. A calculation in which all drip shields are assumed to fail simultaneously at the beginning of the postclosure period increased the peak expected dose by less than 75 percent relative to the nominal scenario, which is still at least 2 orders of magnitude below the regulatory limit. The higher temperature associated with rockfall and the potential increase in the amount of water entering the drift are not expected to significantly increase this result.

Sustained rockfall rubble loads may cause some of the drip shields and waste packages to fail. At the present time, however, the TPA code does not have a model to compute mechanical failure of the waste package if the drip shield is no longer capable of isolating it from rockfall loads. To address this interim limitation, another TPA calculation was performed to bound the potential effects of both the drip shields and waste packages failing at the time of repository closure. The results of this analysis (Figure 4.3.2-1) indicate that a simultaneous failure of all drip shields and waste packages increases dose significantly above the nominal scenario dose, which has a limited number of waste packages (i.e., 40 on average) failing before 10,000 years. However, the dose is below the regulatory dose limit, and the timing and extent of drift collapse is highly uncertain. This uncertainty has implications that the significance associated with the process could range from minimal to high, but is expected to be less than the estimates of this bounding analysis. Additionally, a drip shield design that limits damage to waste packages would limit the significance of drift collapse.

The insulating effects of the rockfall rubble will increase drip-shield, waste-package, and waste-form temperatures. High temperatures will adversely affect the load-bearing capacity of the drip shield and the waste package, thus increasing their failure potential during the duration of high temperatures. The increased temperature also may accelerate drip shield and waste-package corrosion and waste form dissolution.

The DOE calculations with engineered backfill, which can be viewed as an upper-bound for natural backfilling with rockfall rubble, indicates that the peak waste-package temperature could increase nearly two-fold to 315°C, compared to the no-backfill scenario (DOE, 2000; p. 90). However, backfill from drift degradation is anticipated to contain more void space during times relevant to the repository thermal pulse than engineered backfill, thereby, limiting peak waste-package temperature. The temperature of other components of the engineered system could be affected correspondingly. It is unlikely that liquid water will be present within the drift at temperatures above 160°C. The NRC temperature estimates for the unbackfilled condition and the DOE temperature estimates for engineered backfill bound the time for which temperatures are elevated. The analyses estimate that drift degradation is not expected to substantially increase the length of time that EBS components remain above the critical temperature threshold for the occurrence of localized corrosion. Therefore, the effect on corrosion may not be substantially different from the nominal no-backfill scenario. The NRC performed an analyses to determine the effect of increased temperature as a result of backfill from drift degradation on waste form dissolution. The use of a higher-dissolution rate model in which the waste form dissolves in less than 1000 years (e.g., Model 1 in NRC's TPA code) (Mohanty, et al., 2004) as a surrogate for the effect of increased waste-package temperature indicates a 150-percent increase in the peak expected dose relative to the nominal scenario (Figure 4.3.2-2).

<u>Uncertainties</u>

The following are key areas in which uncertainties exist:

- The effects of potential mechanical interactions between the drip shield and waste package under rockfall and seismic conditions are uncertain.

- The effects of drift degradation on water seepage into the drifts are uncertain.

- The effect of elevated temperatures caused by backfill from drift degradation may have an important effect on the creep rate of the drip shield titanium alloys. The impact of the effects are uncertain at this time, however, analyses estimate that drift degradation is not expected to substantially increase the length of time that EBS components remain at significantly elevated temperatures.

- In the case of low-probability intrusive igneous activity event, magma would flow into the drifts because of the pressure gradient between the magma conduit and the drift opening (Woods, et al., 2002). An extensive blockage of the drifts by rockfall rubble may create a barrier against potential magma flow down the drifts, based on analogy with channelized flow of lavas. The extent of blockage necessary to halt a pressure-driven flow is uncertain at this time. In addition, partial backfill of waste packages and drip shields by rubble may create a buffer zone between possible magma and engineered materials, changing potential thermo-mechanical effects from conductive to convective. The possible rubble zone, however, may have elevated temperatures and advect corrosive gasses evolving from the potential magmas (Connor, et al., 1997).

Effects on long-term barrier materials from these processes remain uncertain at this time. See Section 4.3.10.1 for more detailed discussion on the impacts of these uncertainties on magma intrusion into the drifts.

Dynamic Effects of Rockfall on Engineered Barriers: Low Significance to Waste Isolation

The mechanical response of the EBS to discrete dynamic rock block impacts is dependent upon the formation of discrete rock blocks of sufficient size within rock units of the repository horizon. Rock blocks of sufficient size to damage EBS components are only expected for a small portion of the repository.

Discussion

It has been determined that the formation of discrete rock blocks of significant size within the lower lithophysal rock unit is unlikely because of its highly fractured nature (Gute, et al., 2003). The analysis of the middle nonlithophysal rock unit, however, indicated that there are rock blocks of sufficient size to cause damage to the drip shield.

Because the middle nonlithophysal rock unit represents a relatively small percentage of the repository footprint (i.e., less than 30 percent) and only 40 percent of the rock blocks in this unit will have a mass greater than 2.25 metric tons, the potential effect of discrete dynamic rock block impacts on overall repository performance has been determined to be a low-risk scenario for mechanical disruption (Gute, et al., 2003).

Uncertainties

The primary uncertainties associated with assessing the potential effects of dynamic rockfall on the drip shield are the rock-block impact location and impact-velocity distributions. Assuming the rock-block impacts occur at the same location on the drip shield (i.e., the drip-shield crown) with varying impact velocities (corresponding to the calculated fall heights), it was determined that the damage incurred by the drip shields, from this potential disruptive event, is negligible (Gute et al., 2003).

Effects of Seismic Loading on Engineered Barriers: Medium Significance to Waste Isolation

The mechanical response of the EBS and waste form to seismic loading is strongly affected by the extent of accumulated rock fall rubble and associated elevated temperatures.

Discussion

The drip shield and waste package may be breached by the accumulation of damage caused by multiple seismic loading events (in addition to other mechanically disruptive events). It is expected that seismic events will increase the effective static load because of rockfall on the drip shield and waste packages.

Uncertainties

At the present time there is a great deal of uncertainty associated with the seismic threshold loads needed to generate appreciable drip shield and waste-package mechanical damage. These threshold loads are strongly affected by the anticipated in-drift conditions (i.e., the presence of accumulated rockfall rubble and the concomitant elevated temperatures) and the different levels of material and structural degradation that are caused by various corrosion processes; fabrication effects (residual stresses and loss of ductility from welding and quenching); and hydrogen embrittlement.

Effects of Faulting on Engineered Barriers: Low Significance to Waste Isolation

Mechanical disruption of the EBS caused by faulting is dependent upon the mechanical loads placed on the EBS components by fault displacements. Analyses considering patterns of surface-faulting displacements from historical earthquakes in the Basin and Range physiographic province indicate the potential effect of direct fault rupture of EBS components is expected to be small.

Discussion

Analysis of the potential effect of direct, fault rupture on the engineered barrier estimates the risk is small [on the order of tenths of microsieverts (tens of microrems) or less]. The significance of faulting has been evaluated by using the FAULTO module in the TPA Version 4.1j code and by considering the patterns of surface-faulting displacements from historical earthquakes in the Basin and Range physiographic province (Stamatakos, et al., 2003). Both analyses relied on conservative, and in some cased, upper-bound assumptions. For example, no credit was taken for the mechanical strength of the waste packages, because of limited data and analyses on the mechanical behavior or waste packages during a faulting event. All waste packages intersected by faults were assumed to fail. Similarly, because of limited data on the amount of distributed and secondary fault displacements anticipated in the repository during an earthquake at Yucca Mountain, the faulting analyses conservatively assumed that all secondary or distributed faults were capable of failing waste packages, independent of the amount of fault displacement that might occur on the secondary or distributed faults. Nonetheless, risk estimates by both methods are similar and very small as shown in Figure 4.3.2-3. These analyses show that faulting does not contribute significantly to overall repository risk, even with the conservative and upper bound assumptions.

Uncertainties

There are large uncertainties in both the amount and likelihood of fault displacements, and the associated mechanical loads placed on the waste packages and drip shields by these displacements. However, the bounding assumptions employed in the analyses discussed above provide confidence that these uncertainties are still not expected to result in faulting significantly contributing to overall repository risk.

4.3.2.2 References

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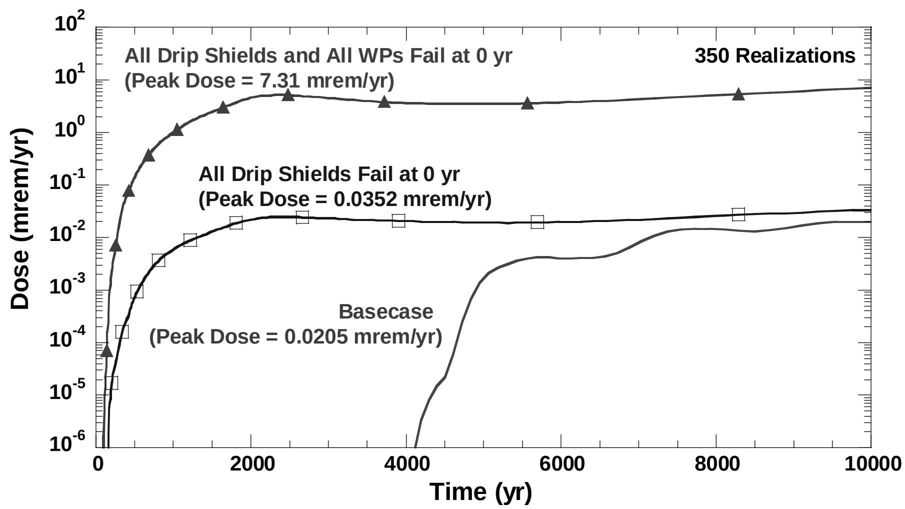

Figure 4.3.2-1. Conditional* peak expected doses corresponding to: (i) the base case; (ii) a hypothetical case in which all drip shields have failed at post-closure; and (iii) a hypothetical case in which all drip shields and waste packages (WPs) have failed at the time of postclosure. (*Doses have not been weighted by the probability of scenario occurrence.)

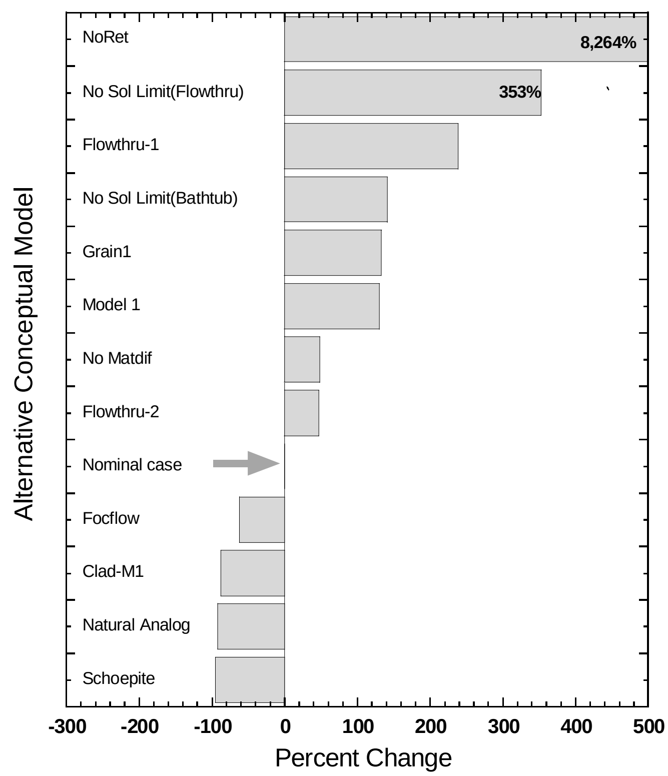

Figure 4.3.2-2. Effects of alternative conceptual models at 10,000 years. (From Mohanty, et al., 2004; Figure 4-15)

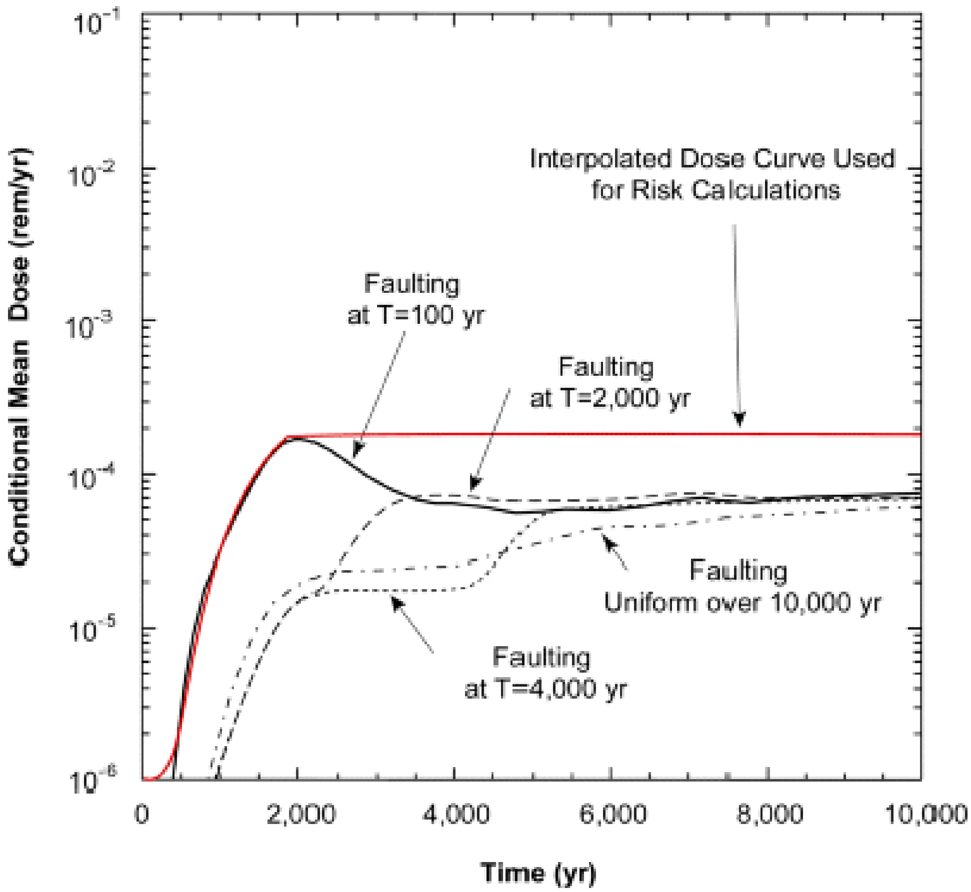

Figure 4.3.2-3. Comparison of the mean dose versus time curves for cases in which the faulting events on the Solitario Canyon Fault were forced to occur in all 500 realizations at a specified year (100, 2000, and 4000); and for a case in which the faulting events occurred uniformly over the entire 10,000-year post-closure period.

4.3.3 Quantity and Chemistry of Water Contacting Waste Packages and Waste Form (ENG3)

Risk Insights:	
Chemistry of Seepage Water	High Significance

4.3.3.1 Discussion of the Risk Insights

Chemistry of Seepage Water: High Significance to Waste Isolation

Evaluating the range in chemistry of water seeping into the drift and contacting the drip shield and waste package is important for determining corrosion rates of the engineered materials.

Discussion

The quantity and chemistry of water that drips onto drip shield/waste package surfaces largely determine the composition of the salts that can precipitate on those surfaces. The composition of a salt determines the composition of its associated brine, and the relative humidity conditions under which that salt will deliquesce to form a brine. The composition of the salt is largely determined by the chemistry of the evaporating water, whereas the timing and extent of salt formation/evaporation is largely determined by thermal-hydrologic conditions in and above the repository over time. Changes in seepage rate and the conditions appropriate for evaporation and salt formation depend on diverse factors, such as climate, flow pathways in the UZ, heat load, drift design, and local controls on flow pathways (i.e., roughness of drift crown surface, the location of rock bolts and other engineered materials, and the timing and extent of rock fall and drift degradation).

The expected lifetimes of the drip shield and waste package depend strongly on the chemistry of water contacting their surfaces. Low pH and elevated concentrations of certain dissolved anionic species are expected to enhance aqueous corrosion—promoting specific corrosion modes, either uniform corrosion, or localized corrosion, the latter of which has orders-of-magnitude higher rates than uniform corrosion. For example, the susceptibility to localized corrosion of Alloy 22, the material the DOE plans to use for the outer container of the waste package, is strongly dependent on pH and on chloride ion concentration (Dunn, et al., 2003). Titanium alloy grade 7, the material planned to be used for drip shields that would cover the waste packages, has been shown to be susceptible to generalized corrosion from fluoride ions (Brossia, et al., 2001), and other data indicate that titanium is susceptible to localized corrosion in the presence of bromide ions (Beck, 1971). On the other hand, other anionic species can inhibit localized corrosion if present in sufficient concentration. For example, nitrate ions have been shown to inhibit localized corrosion of Alloy 22 in chloride solutions (Dunn, et al., 2003). However, titanium corrosion may be limited by the supply of fluoride from dripping water and not strictly to the concentration threshold (Lin, et al., 2003).

High concentrations of these anionic species could result from the evaporation of seepage waters, even if those waters initially were dilute (Brossia, et al., 2001). Evaporation experiments (Rosenberg, et al., 2001) and some limited thermodynamic analyses (Pabalan, et al., 2002a,b) show that waters of widely varying initial composition evolve by evaporation toward three types of brines: (1) calcium-chloride (Ca-Cl); (2) neutral; and (3) alkaline brines. The results of thermodynamic simulations of evaporation of Yucca Mountain ground waters indicate that Ca-Cl brines tend to have chloride to nitrate ratios sufficiently high to make the corrosion-inhibiting property of nitrate ions ineffective (Pabalan, et al., 2002a,b). The thermodynamic calculations also show that alkaline brines tend to have fluoride ion concentrations that exceed 0.0005 moles/kg H_2O, a threshold fluoride concentration at which titanium grade 7 has been observed to undergo accelerated general corrosion in deareated 1-M sodium chloride solutions (pH 6.4 to 8.2) at 95°C (Brossia, et al., 2001). The thermodynamic analyses indicate that the three brine types have elevated bromide concentrations, but the potential effect of the elevated bromide concentrations is uncertain because of the limited data on corrosion of titanium in bromide solutions.

In addition to elevated concentrations of anionic species, evaporation of seepage waters possibly could lead to acidic condensates that could enhance the degradation of engineered materials. For example, experiments by Pulvirenti, et al. (2003) demonstrated a high corrosion rate for Alloy 22 that was reacted at 130°C with acidic condensate formed by evaporation of simulated Yucca Mountain brines. The fast corrosion rate was interpreted as resulting from the formation of hydrochloric acid and nitric acid during evaporation of the highly concentrated brines. The formation of hydrochloric and nitric acid was supported by chemical tests of vapor samples in a closed system. While comparable temperatures are expected for the repository, the open nature of the Yucca Mountain system (i.e., the movement of air in the unsaturated zone), and the buffering effect due to minerals in the host rock, significantly limit the concentration of hydrochloric and nitric acid.

Additional salts, such as soluble chlorides, nitrates, and sulfates of sodium, potassium, calcium, and magnesium, that are present as aerosols in atmospheric air (Ge, et al., 1998) and entrained in ventilation air introduced into the repository drift also may be deposited on the drip shield and waste package surfaces (BSC, 2001). Inorganic salts are generally hygroscopic and will absorb moisture from humid air, generating small volumes of potentially corrosive brines. A phase change from a solid particle to a saturated aqueous phase usually occurs spontaneously when the relative humidity in the surrounding atmosphere increases to a level known as the deliquescence point or deliquescence relative humidity (DRH) (Tang and Munkelwitz, 1993). The DRH of salt mixtures depends on the mixture composition and temperature. Experimental data and thermodynamic analyses (Pabalan, et al., 2002a,b) show that mixtures containing magnesium and calcium have much lower DRH than those containing sodium and potassium. If magnesium and calcium salts are deposited on waste-package and drip-shield surfaces, deliquescence and brine formation could occur when the repository temperature is above 100°C and could lead to early initiation of aqueous corrosion of the drip shield and waste-package engineered barrier.

Uncertainty

Some uncertainty arises from the uncertainty in the quantity of water entering the drift and the timing of and the temperature during which seepage occurs. Published experimental and modeling studies of evaporation have implicitly assumed that seepage water is present and that the temperature is sufficiently high to cause evaporation. However, the flux of seepage water over time depends on diverse factors, such as climate, flow pathways in the UZ, heat load, drift design, and local controls on flow pathways (i.e., roughness of drift crown surface, the location of rock bolts and other engineered materials, and the timing and extent of rock fall and drift degradation).

The most significant source of uncertainty in determining the chemical environment for corrosion is the range of in-drift water compositions that may result from spatial and temporal variations in seepage water composition, the composition and amount of condensed water formed by cold-trap processes, and the extent of chemical interactions between these waters and engineered and natural materials. Coupled thermal-hydrological-chemical processes occurring in the rocks that overlie the proposed repository will largely determine the quantity and chemistry of water seeping into the drifts. Explicit and comprehensive evaluation of these coupled processes, however, generally requires the construction of reactive transport models (CRWMS M&O, 2001).

Reactive transport predictions have three substantial sources of uncertainty that are difficult to quantify (Browning, et al., 2003a,b): (1) code limitations, especially algorithms that reflect a representation of coupled thermal-hydrological-chemical processes; and (2) uncertain parameter values. A comprehensive evaluation of the uncertainties associated with conceptual models and code limitations requires an assessment of how omitted processes and components affect model results. It is generally not feasible to quantify precisely the uncertainties associated with the myriad individual input parameters and constraints typically associated with reactive transport models, limiting assessments to analyzing the effects for a range of potential chemistries. Nevertheless, these uncertainties are constrained by the system characteristics (e.g., relatively homogeneous rock chemistry) and by the well understood dynamics of evaporation (i.e., the chemical divide phenomenon results in a diversity of water compositions evolving to a few brines).

The composition of in-drift water is a significant factor in determining the types of salts that can form and the corrosiveness of the associated brine. Bounding the range of in-drift water compositions requires characterization of diverse in-drift features, events, and processes (FEPs) that may alter the chemical composition of seepage waters. As noted above, these FEPs include cold-trap processes that control the timing, locations, and compositions of condensed waters; the development of preferential flow pathways that may focus flow over drip shield/waste package surfaces; the chemical consequences of degradation of the engineered drifts; and the chemical evolution of in-drift waters that have interacted chemically with engineered materials. The impact of these FEPs on the quantity and chemistry of in-drift water is believed to be of less importance.

Some uncertainty also remains with respect to the composition of dust that may be deposited on the drip-shield and waste-package surfaces, and the extent to which chemical interactions between dust and in-drift waters may alter the chemical environment for corrosion. Chemical analyses by the U.S. Geological Survey (Peterman, et al., 2003) on a limited set of dust samples from the exploratory shaft facility (ESF) indicates that a large proportion of the dust in the ESF comes from comminution of the rock during tunnel construction and autogenous grinding of muck during haulage on the conveyor belt. However, the U.S. Geologic Survey data also indicate that salts of evaporated construction and native pore water are minor components of the dust. There are limited data to support extrapolation of the composition of dust sampled from the ESF to that of dust that may be encountered in the repository drift. There is also uncertainty with respect to the deliquescence behavior of salts mixed with rock dusts.

4.3.3.2 References

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4.3.4 Radionuclide Release Rates and Solubility Limits (ENG4)

Risk Insights:	
Waste Form Degradation Rate	Medium Significance
Cladding Degradation	Medium Significance
Solubility Limits	Medium Significance
Mode of Release from Waste Package	Low Significance
Effect of Colloids on Waste Package Releases	Medium Significance
Invert Flow and Transport	Low Significance
Criticality	Low Significance

4.3.4.1 Discussion of the Risk Insights

Waste Form Degradation Rate: Medium Significance to Waste Isolation

The dissolution rate of the waste form in an aqueous environment is important for all radionuclides. Uncertainty in the dissolution rate is large such that the time required to release radionuclides from the spent-fuel matrix or vitrified-waste forms can vary from hundreds of years to hundreds of thousands of years. Water chemistry and temperature within the waste package could affect the degradation rate of the spent fuel and vitrified wastes. Corrosion of the internal metallic components of the waste package (e.g., fuel assembly baskets) could reduce pH, leading to higher dissolution rates from spent fuel and vitrified wastes.

<u>Discussion</u>

Several studies point to the sensitivity of dose to the dissolution rate of the spent fuel source material. The pre-exponential coefficient for the base case spent nuclear fuel dissolution model (exponential dissolution Model 2) in the TPA code is one of the most sensitive parameters (Table 4.3.4-1) (Mohanty, et al., 2004). Furthermore, among the four alternative models for spent-fuel degradation in TPA Version 4.1 (which have markedly different release rates), there is a clear correlation between release rate and dose (Figure 4.3.4-1). Dissolution rate is a relatively important determinant of repository performance, but the overall peak dose is not directly proportional to it, because other mechanisms like diffusion, advection, and solubility attenuate ultimate release rates from the engineered barrier. The previous DOE total system performance assessment (TSPA) models use fuel dissolution rates considerably larger than those used by the NRC (CRWMS M&O, 2000). Using more aggressive chemical conditions (Model 1) in place of the base case (Model 2) leads to a two-order of magnitude increase in release rate, yet, as shown in Figure 4.3.4-1, the increase in peak expected dose is only a factor of approximately 2.5. The NRC staff expects the base case (Model 2) to represent chemical conditions to which spent fuel could be exposed under anticipated repository conditions.

Uncertainties

The dissolution rate of spent fuel depends on temperature, the presence of oxygen, and the chemistry of in-package water. There are similar controls on the dissolution of vitrified waste, although it is not significantly affected by oxidation potential. Furthermore, the rate at which radionuclides can leave the waste package may be restricted by: (1) the amount of water coming into contact with the waste form and escaping the waste package; (2) diffusion processes inside and outside the waste package; and (3) solubility, and not strictly by the dissolution rate of the fuel. Model 4 (schoepite dissolution model) recognizes that the spent fuel can degrade relatively quickly in the presence of water and oxygen, but that some radionuclides may be incorporated into secondary minerals formed during spent fuel degradation and released at a slower rate by subsequent dissolution of these less-soluble phases.

Cladding Degradation: Medium Significance to Waste Isolation

Zircaloy cladding exhibits extremely low uniform corrosion rates in aqueous environments and could delay substantially the release of radionuclides from commercial spent fuel if it remains intact. Performance assessments estimate a high correlation between dose and fraction of failed cladding. However, cladding is thin and not physically strong. Cladding failure can occur as a result of localized corrosion, stress corrosion cracking, and hydride reorientation, under a combination of adverse environmental and stress conditions. Cladding may also fail as a result of creep caused by hoop stresses due to internal pressure, or mechanically, when subjected to loads associated with seismic events and rockfall.

Discussion

The DOE has considered that cladding can be an effective metallic barrier against the release of radionuclides from commercial spent fuel. However, little experimental evidence has been provided to support such assessment nor have solid technical bases for the assumptions included in the model abstraction been developed. This is the case, in particular, for the modeling of localized corrosion and stress corrosion cracking, as well as for the lack of consideration of hydride reorientation as a potential failure process that may be faster and, hence, more detrimental than unzipping alone. Recently, the DOE has provided performance assessment calculations showing that the cladding degradation rate at the 95th percentile, or even complete neutralization, only increases the mean dose by one order of magnitude, and the dose is more than four orders of magnitude lower than that specified in the regulations at 10 CFR Part 63 for the RMEI (Figure 4.3.4-2). However, in their nominal case, the DOE assumes that the fraction of failed cladding perforated before unzipping remains constant at 0.08 during the 10,000 yr compliance period, reaching only 0.2 after 100,000 yr (CRWMS M&O, 2000). Also, these estimates of cladding protection do not consider the full range of possible failure mechanisms or their probabilities, and therefore may overestimate the effectiveness of cladding as a barrier. In other international programs (e.g., France, Canada, Sweden, Finland), no credit is given to the cladding as a metallic barrier as a result of the uncertainties related to its integrity under repository conditions.

It is apparent in the TPA Version 4.1 Sensitivity Report (Figure 4.3.4-3) (Mohanty, et al., 2004) that the introduction of cladding protection decreases the dose at 10,000 yr with respect to that for the base case from 2×10^{-4} to 3×10^{-6} mSv/yr (2×10^{-2} to 3×10^{-4} mrem/yr). Release rates of highly soluble and mobile radionuclides like Tc-99 and I-129 account for most of the 10,000-

year predicted dose, and are approximately proportional to the amount of spent fuel exposed. Other hazardous but less mobile radionuclidess like plutonium and americium may not be as affected by the amount of spent fuel exposed, because their release is likely to be controlled by solubility limits.

Uncertainties

Cladding perforations may expose increasing areas of the spent fuel matrix to the in-package environment as a result of splitting (unzipping) caused by the oxidation of UO_2 pellets, either by air and water vapor or by liquid water. Perforations could exist in the cladding before waste emplacement and can develop by the corrosion and mechanical processes described above. Considerable uncertainties exist regarding the initial stage of cladding failure during which perforations occur, particularly in the case of corrosion processes. These uncertainties are mostly associated with uncertainties in the in-package environment and the initial condition of the spent fuel. A more significant range of uncertainties can be expected for high burn-up fuel. Another concern is the potential for cladding failure under seismic conditions, rockfall, and drift collapse, with their associated uncertainties.

Solubility Limits: Medium Significance to Waste Isolation

Solubility limits effectively limit the release of many radionuclides.

Discussion

Solubility limits can be important factors in the release of radionuclides from the engineered barrier and ultimately to dose. Figure 4.3.4-4 shows the sensitivity of the DOE's calculated doses to neptunium solubility. The degree to which radionuclide release to the environment is solubility-limited depends on the intricate interaction among the waste-form leaching rate, the degree of waste-form exposure to water, radionuclide half-life, radionuclide inventory, and the position of the radionuclide in the decay chain. Of the three radionuclides estimated to be major contributors to dose in a recent numerical study (i.e., Tc-99, I-129, and Np-237), only the release of Np-237 would be significantly decreased by its solubility limit (Mohanty, et al., 2003). However, 13 out of 20 radionuclides considered in this study exhibited solubility-limited behavior. Figure 4.3.4-5 plots the difference in cumulative release for each radionuclide from the waste packages between the base case and a modified case in which the solubility limits for all radionuclides were set to an artificially high value. This figure shows that radionuclidess such as americium and plutonium exhibit increased release when solubility limits are increased. Under the expected performance of other barriers, releases of radionuclidess like plutonium and americium would not normally be seen because they are relatively immobile in the geosphere.

Uncertainties

The solubility value for a particular radionuclides used in performance assessment calculations will depend on assumptions regarding the solubility controlling solid phase for the radionuclides of concern. The assumed solid phase may differ in its solubilities by several orders of magnitude. For some radionuclidess, such as neptunium, incorporation into a low-solubility secondary phase is possible, but the evidence for this mechanism can be tenuous (Fortner, et al., 2003).

A necessary assumption in assigning solubility values pertains to the physical-chemical conditions of the system. The solubility of a radionuclide will depend on the composition of the aqueous phase as well as on its temperature and oxidation state. Inorganic and organic ligands that can form aqueous complexes with the radionuclide may be present. Complexation increases the amount of the radionuclide in the solution for elements such as uranium, neptunium, plutonium, and americium. Actinide solution chemistry in environmental waters would be dominated by hydroxide and carbonate complexation; thus, the solubility of actinide solids would be highly dependent on pH, aqueous carbonate concentration, and partial pressure of carbon dioxide gas. The solubility of some radionuclides depends strongly on their oxidation states.

In some cases, radionuclides released from the waste form may become sorbed to solid materials. This sorption could either hinder or enhance transport from the engineered barrier, depending on whether the substrate were immobile or mobile. In the former case, experiments with degrading spent fuel show a large fraction of plutonium and americium releases became attached to the container walls (Wilson, 1989). In the latter case, radionuclides that become attached to colloids (e.g., hydrous ferric oxides generated by the corrosion of steel in the waste packages and ground support) might allow release of radionuclides in amounts greater than estimates based on solubility limits alone.

Mode of Release from Waste Package: Low Significance to Waste Isolation

The significance of diffusional release will depend on a number of assumptions such as water-film thicknesses, diffusion distances inside and outside the waste packages, unclogged openings, and a mechanism to sweep away contaminants, to keep the concentration gradients high. Advective releases rely on the quantity of water entering and leaving the waste package, and could be more significant than diffusion after degradation processes cause openings in the waste package that are sufficiently large to allow dripping water to come into direct contact with the waste. Processes leading to openings in the waste packages include localized corrosion, stress corrosion cracking coupled with mechanical loading events from dynamic and static rock-fall loads, and intrusive igneous activity disruptive events.

Discussion

Scenarios capable of allowing diffusion in amounts that would be sufficient to cause significant releases are highly unlikely. There would have to be a continuous and substantial water pathway for diffusion and a mechanism to keep the concentration gradient high. Water films are likely to be thin or discontinuous. Under conditions where only water vapor is present in the drift, water films inside the waste package would be limited to layer thicknesses measured in tens of molecules or less. Mechanisms for flushing diffused radionuclides away from the waste package (i.e., to keep the concentration gradient high) are unlikely, requiring liquid water to drip onto cracks or holes in the waste package. Calculations of diffusion under any likely condition show that releases by this mechanism are unlikely to cause doses anywhere approaching the dose limits, even under the failure or underperformance of other barriers.

Uncertainties

Mechanisms whereby the rock is forced into close contact with the waste packages may enhance diffusion by lowering the external resistance. Such processes may include engineered backfill, infilling of drifts by collapse, and igneous intrusion.

Degradation of the waste package that result in large openings and short diffusion pathways inside the waste package would reduce internal resistance to diffusion.

Effect of Colloids on Waste Package Releases: Medium Significance to Waste Isolation

Colloids can form from the aqueous degradation of fuel and especially vitrified waste. For the degradation of fuel, dissolved radionuclides, in addition to colloids directly resulting from waste form degradation, can attach to natural or anthropogenic (man-caused) colloids, especially iron oxyhydroxides formed from corrosion of the steel in the waste package and ground-support materials. Degradation of glass in vitrified waste can form clay colloids such as smectite and illite, which can also be the substrate for radionuclide attachment. Colloids can be transported out of the waste package primarily by advection in flowing water. However, colloids may be easily filtered once in a porous medium such as the invert, the Calico Hills vitric unit, and the alluvium.

Discussion

Colloids may be important to repository performance if sufficient experimental evidence indicates they will form in large amounts and not be substantially removed in the subsurface through coagulation and filtration. Figure 4.3.4-6 shows the results of a bounding approximation of colloid effects by eliminating all retardation of plutonium, americium, and thorium isotopes. These results led to significant increases in dose, although still below the 0.15 mSv/yr (15 mrem/yr) regulatory criterion (Contardi, et al., 1999; Mohanty, et al., 2003). However, these results were highly conservative because they assumed that all releases from the waste form would leave the waste package and not be removed by filtration or straining. Experimental results on plutonium and americium releases from spent fuel suggest that colloidal radionuclides easily become attached to surfaces such as the interior of the waste package, and may not be released easily (Wilson, 1989).

The DOE has performed analyses to evaluate the sensitivity of colloids where plutonium and americium are irreversibly sorbed to waste form colloids (Figure 4.3.4-7). The DOE analyses indicated that colloidal concentrations were only significant in 10,000 years, under an intrusive igneous event, wherein a large number of waste packages are significantly damaged. The potential importance of colloidal transport, if all waste packages were to fail within 10,000 years, is evident in the DOE TSPA-SR (CRWMS M&O, 2000), which indicated colloidal plutonium is the second highest dose contributor after 70,000 years, when a significant number of waste packages are estimated to have failed, allowing release of radionuclides.

Uncertainties

Some measurements in the field indicate that colloids can travel relatively easily in the ground under the right conditions (Kersting, et al., 1999). However, release and transport parameters for colloid effects for the proposed repository are not supported by extensive field and laboratory data and near-field effects are not well-constrained.

Invert Flow and Transport: Low Significance to Waste Isolation

The invert has a short travel pathway relative to the geologic barriers and is not expected to have a significant effect on radionuclide transport in the aqueous phase.

Discussion

Although the invert is likely to consist of a porous or crushed rock material and have desirable properties for radionuclide sorption and possibly colloid filtration, it is very thin compared to other porous materials in the pathway of radionuclide transport such as the Calico Hills vitric unit and alluvium. Performance assessment studies showed practically no effect of eliminating the invert as a barrier (Mohanty, et al., 2004).

Uncertainties

Any benefit of the invert may be lost over time if precipitation of minerals from ground water or alteration of minerals in the material by heat causes the porosity to decrease, thereby allowing short-circuiting around the porous material.

Criticality: Low Significance to Waste Isolation

The potential for criticality to occur either within the waste package or in the geosphere is considered unlikely; in addition, if it were to occur, the consequences would be limited (e.g., at most doubling of the inventory of fission products and locally increasing the temperature).

Discussion

Commercial spent fuel that would be stored in the repository cannot become critical unless there is sufficient water or other neutron moderator material in the waste package, and criticality controls (e.g., poisons) are removed or rendered ineffective. Neither the drip shield nor the waste package are expected to fail within 10,000 years. Furthermore, their failure is not expected to result in water sufficiently filling the waste package to submerge fuel elements. Criticality is also limited by the presence of nuclear poisons such as the borated stainless steel fuel baskets and the actinides and fission products within spent fuel. The boron in the stainless steel may eventually leach out, depending on the corrosion rate of the material and the rate of water circulation through the waste package to carry it away.

Should a criticality eventually occur, the event has been postulated to be either steady-state or transient in nature. In the steady-state case, power from the nuclear chain reaction could be limited by the availability of the water moderator, which would be controlled by the balance of heat generation and removal of heat by conduction and evaporation. The consequences of this situation would be modest, leading to elevated temperature of the waste package, and the generation of additional radioactive inventory in the spent fuel (Figure 4.3.4-8). However, steady-state criticality events may extend for thousands of years at low power levels provided that the waste package remains filled with water. Transient criticality could occur if there were rapid reactivity insertion caused by, for example, the sudden rearrangement of fuel and the water moderator from seismic shaking of partially failed waste packages, or seismically induced sloshing providing more moderation to the less-burned ends of the fuel rods. A transient criticality could potentially disrupt the waste form, cladding, and the waste package through a steam explosion (Figure 4.3.4-9).

Criticality outside of the waste package is not expected because there are no likely mechanisms that could reasonably reconcentrate the released fissile materials into a critical configuration.

Uncertainties

The identification of scenarios resulting in a significant number (greater than 40) of simultaneously long-lived critical waste packages or transient criticality would merit further refinement of the consequence evaluation only if they could be shown to be likely to occur.

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Table 4.3.4-1. Influential Parameters for the 10,000-Year Simulation Period from Sensitivity Analysis Studies. (From Mohanty, et al., 2004, Table 4-10)

No.	Parameter Name	Score
1	Areal average mean annual infiltration at start	7/7
2	Drip shield failure time	7/7
3	Pre-exponential term for spent nuclear fuel dissolution (Model 2)	6/7
4	Subarea wet fraction	6/7
5	Waste package flow multiplication factor	6/7
6	Well pumping rate at 20-kilometer [12.4-mile] receptor group	6/7
7	Fraction of condensate toward repository	4/7
8	Alluvium retardation coefficient for neptunium-237	3/7
9	Distance to Tuff-alluvium interface	3/7
10	Waste package initially defective fraction	3/7

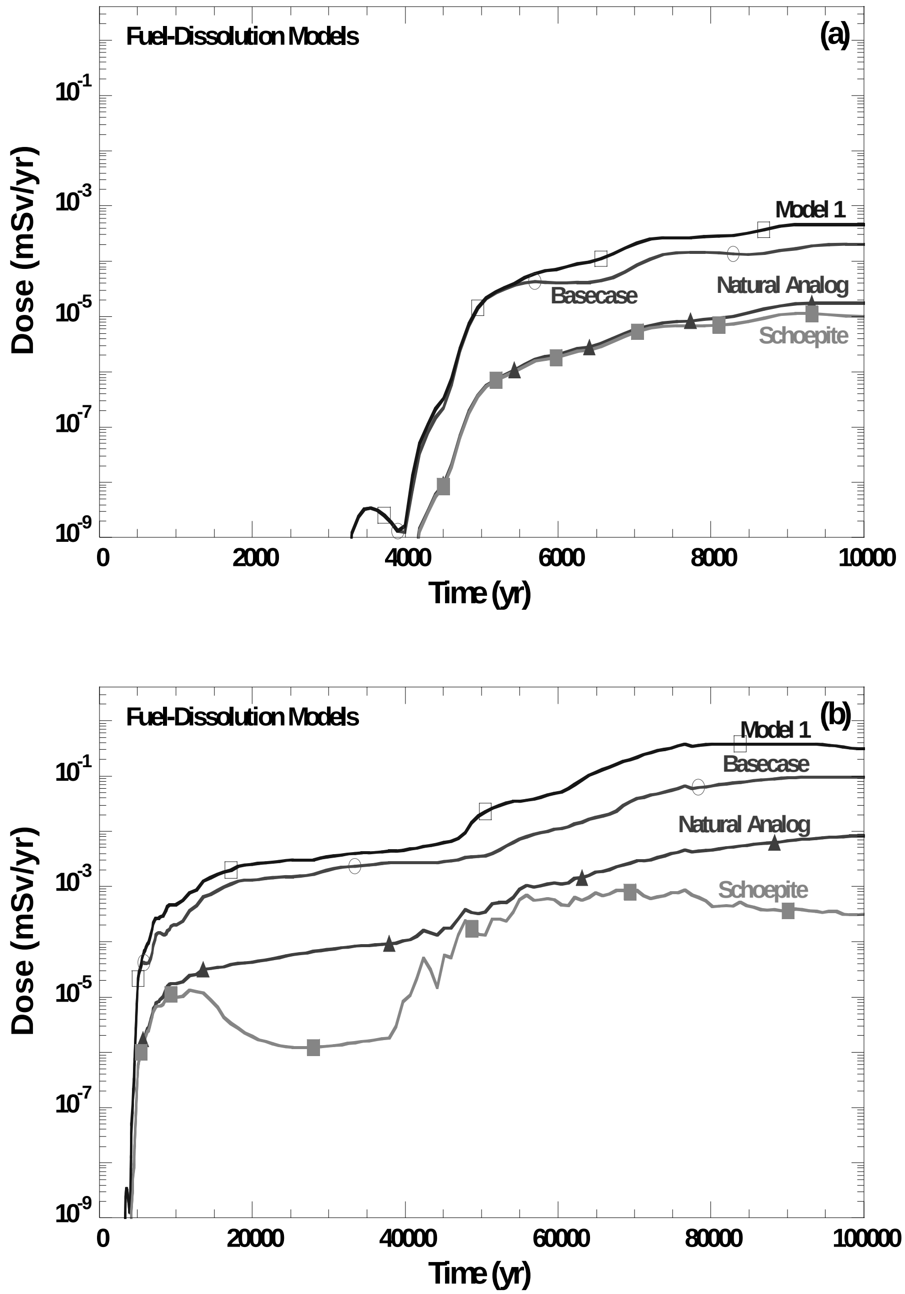


Figure 4.3.4-1. Ground water dose from the base case and the fuel-dissolution alternative conceptual models for: (a) 10,000 years; and (b) 100,000 years, using the mean value data set. (From Mohanty, et al., 2004, Figure 3-36)

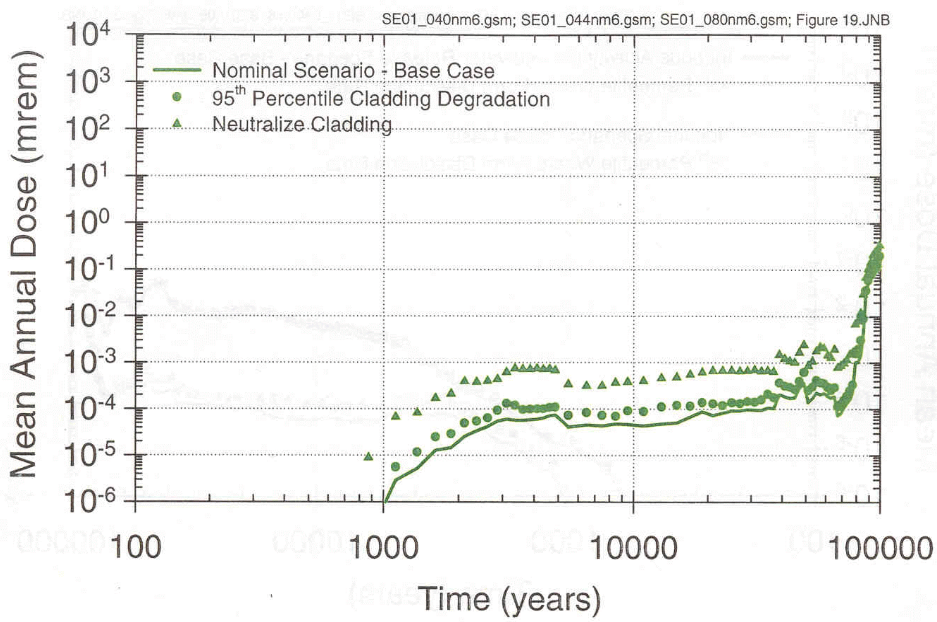


Figure 4.3.4-2. Sensitivity of mean annual dose to the commercial spent nuclear fuel cladding degradation rate. (From BSC, 2002. Note that dose estimates for variations from the base case do not represent variations in expected risk because the probability of the variation is not accounted for.)

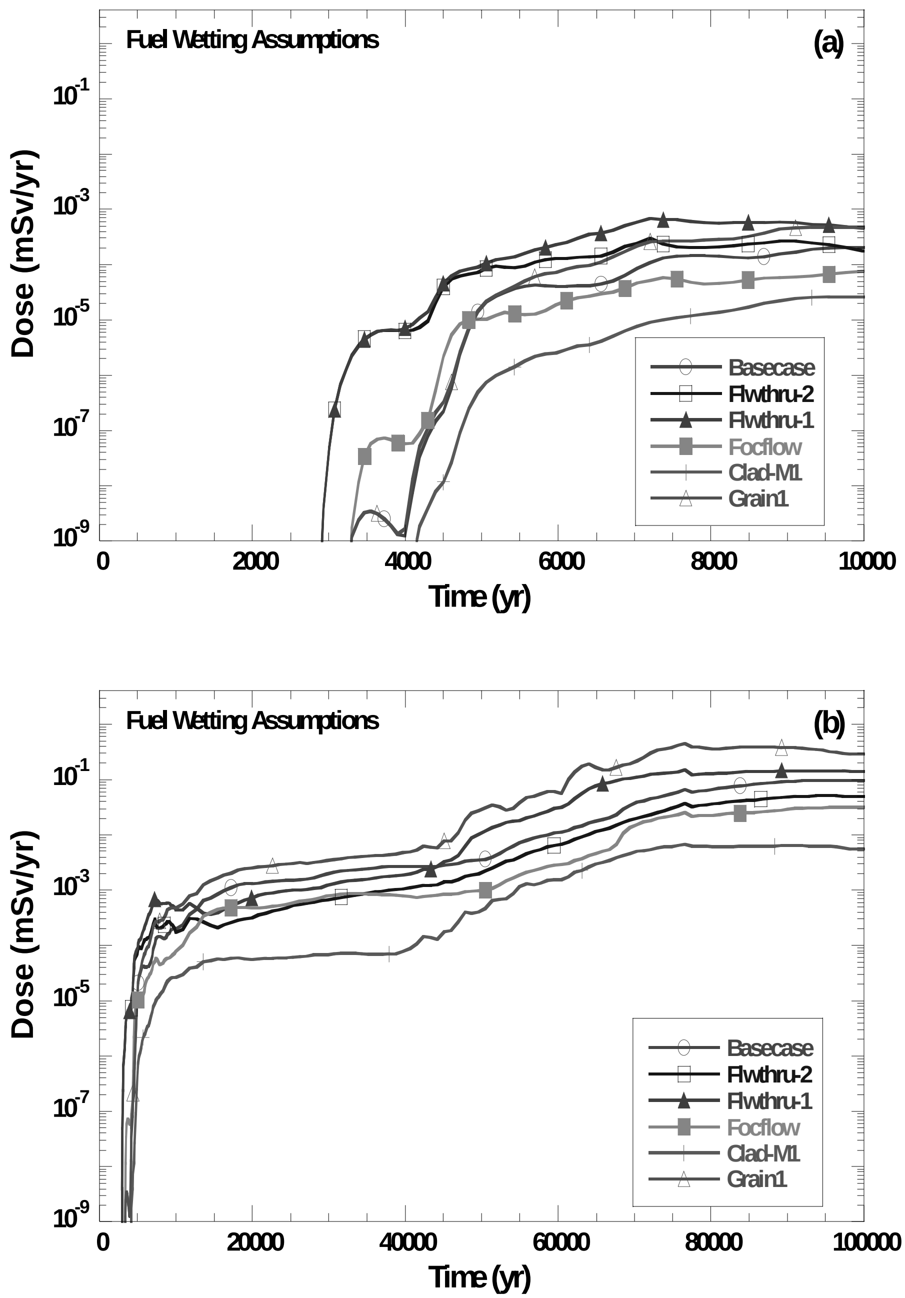


Figure 4.3.4-3. Effects of cladding credit: Ground water dose from the base case and the fuel-wetting alternative conceptual models for: (a) 10,000 years; and (b) 100,000 years, using the mean value data set. (From Mohanty, et al., 2004, Figure 3-37)

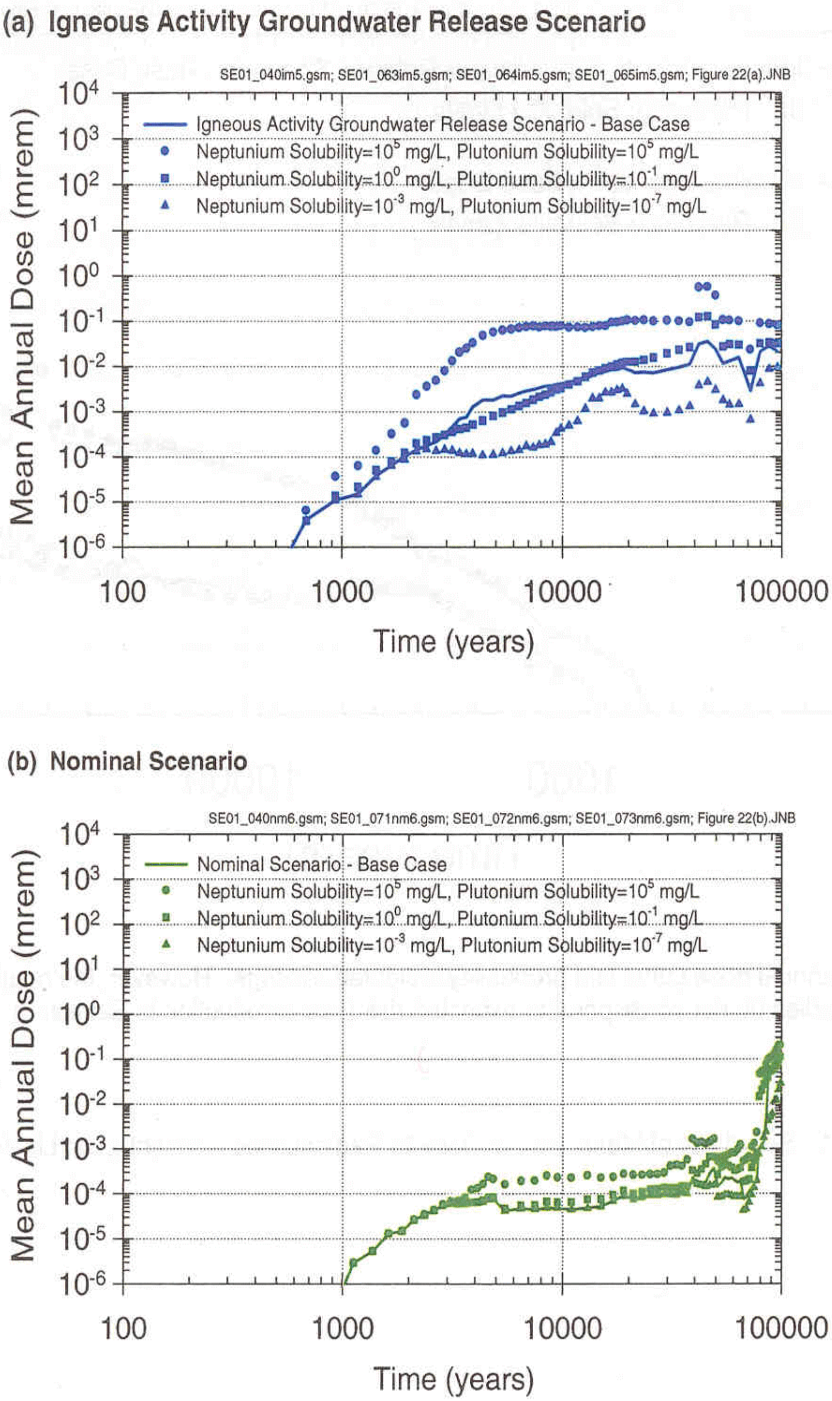


Figure 4.3.4-4. Sensitivity of mean annual dose to concentration limits for neptunium and plutonium.
(From BSC, 2002. Note that dose estimates for variations from the base case do not represent variations in expected risk because the probability of the variation is not accounted for.)

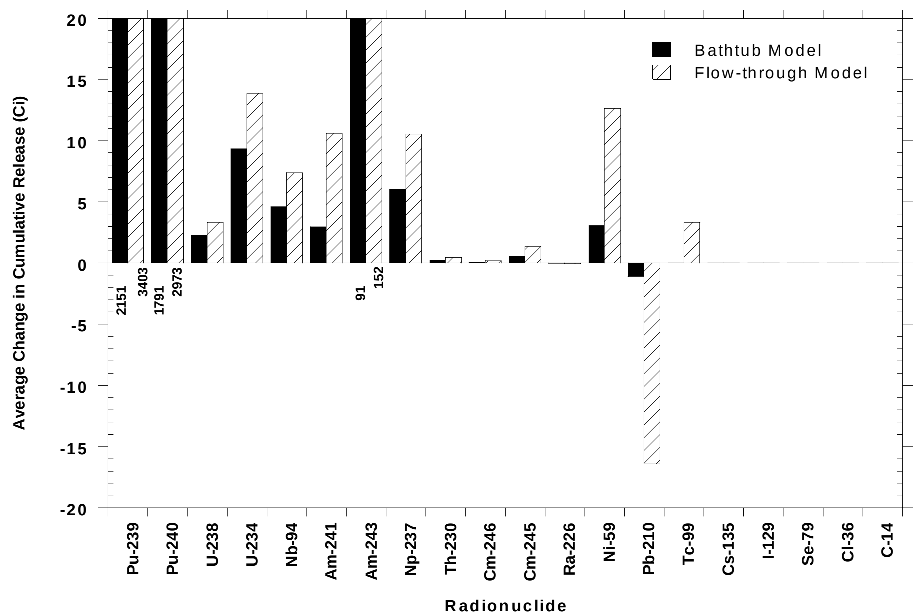


Figure 4.3.4-5. Average change to the nominal scenario cumulative radionuclide release in 10,000 yr when the solubility limit is specified as ineffective in TPA 4.1 simulations (i.e., set to an artificially high value). (From Mohanty, et al., submitted)

Note:

Pu-239	**plutonium-239**	**Cm-245**	**curium-245**
Pu-240	**plutonium-240**	**Ra-226**	**radium-226**
U-238	**uranium-238**	**Ni-59**	**nickel-59**
U-234	**uranium-234**	**Pb-210**	**lead-210**
Nb-94	**niobium-94**	**Tc-99**	**technetium-99**
Am-241	**americium-241**	**Cs-135**	**cesium-135**
Am-243	**americium-243**	**I-129**	**iodine-129**
Np-237	**neptunium-237**	**Se-79**	**selenium-79**
Th-230	**thorium-230**	**Cl-36**	**chlorine-36**
Cm-246	**curium-246**	**C-14**	**carbon-14**

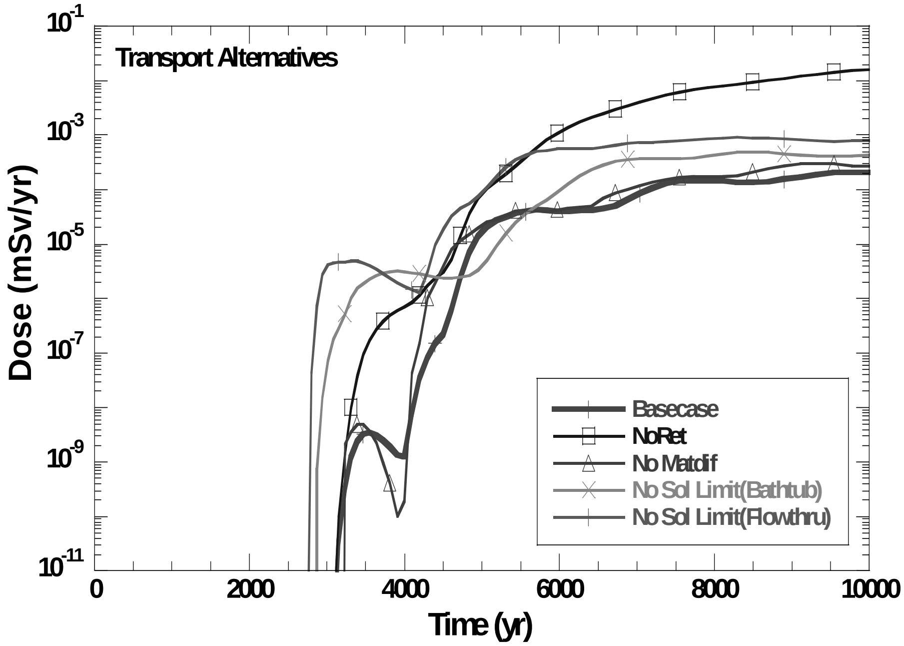


Figure 4.3.4-6. Dose curves for alternative transport models in TPA 4.1 [base case model (Basecase); assumption of no retardation (NoRet); assumption of no matrix diffusion (No Matdif); and assumption of no solubility limits (No Sol Limit) with the bathtub and flow-through models]. The NoRet curve provides a conservative upper bound to colloidal effects simply by assuming no retardation of plutonium, americium, and thorium isotopes. The NoRet case yields doses about one order of magnitude higher than the Basecase, which is less than 10^{-3} mSv/yr (10^{-1} mrem/yr).
(From Mohanty, et al., 2004)

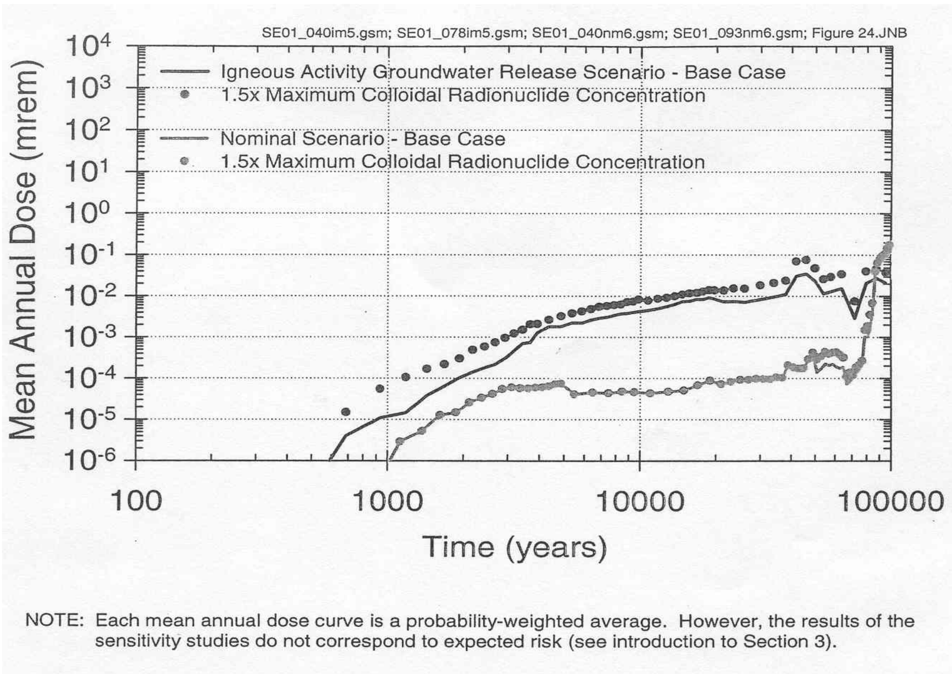


Figure 4.3.4-7. Sensitivity of mean annual dose to concentration of plutonium and americium irreversibly sorbed to waste form colloids.
(From BSC, 2002, Figure 24. Note that dose estimates for variations from the base case do not represent variations in expected risk because the probability of the variation is not accounted for.)

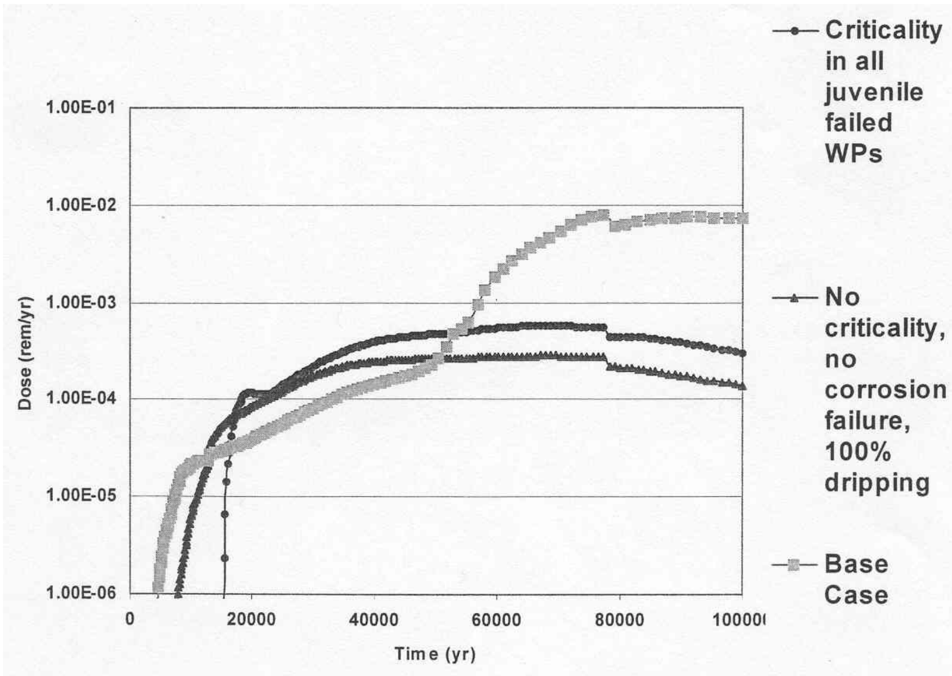


Figure 4.3.4-8. Dose consequence of in-package steady-state criticality [waste packages (WPs)]. (From Mohanty, et al., 2004, Figure G-2)

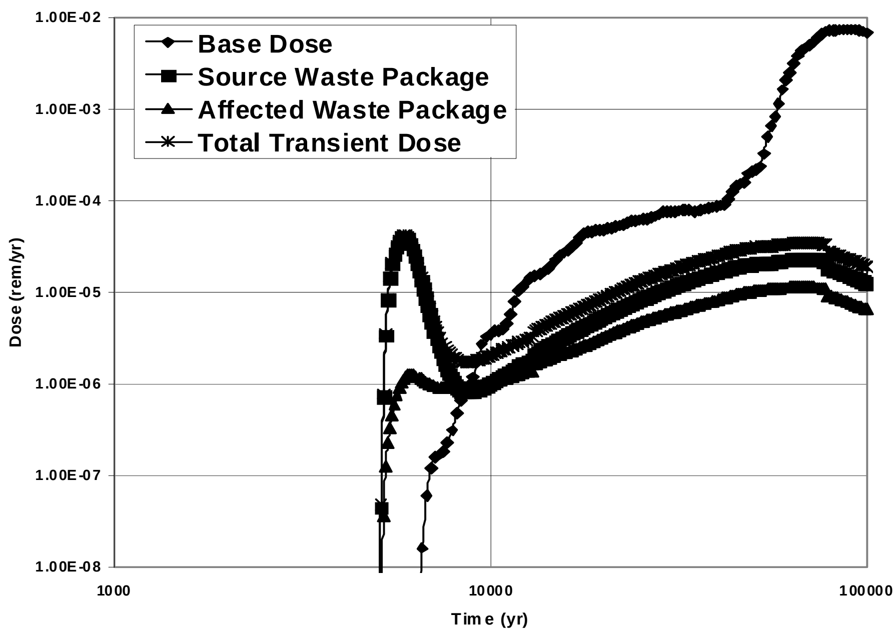


Figure 4.3.4-9. Dose consequence of in-package transient criticality. (From Mohanty, et al., 2004, Figure G-3)

4.3.5 Climate and Infiltration (UZ1)

Risk Insights:	
Present-day Net Infiltration Rate	Medium Significance
Long-term Climatic Change	Medium Significance

4.3.5.1 Discussion of the Risk Insights

Present-day Net Infiltration Rate: Medium Significance to Waste Isolation

Estimates of present-day net infiltration rates are important for estimating the deep percolation rate. The deep percolation rate, in turn, affects the quantity of water coming into contact with the waste packages and waste form.

<u>Discussion</u>

Some of the precipitation that falls on Yucca Mountain is expected to move into the bedrock as net infiltration. Estimates of present-day net infiltration rates are used to directly estimate deep percolation rate at the repository horizon, assuming no lateral diversion of flow. Some fraction of this deep percolation is expected to seep into the repository drifts and come into contact with the waste packages, and, potentially, the waste form. Water coming into contact with waste packages will likely affect the integrity of the waste packages and the release of radionuclides from the waste form. The quantity of water has a more significant effect on the rate of release of radionuclides that have lower solubility limits. Of these radionuclides, Np-237 has the greatest potential to contribute to dose during the period of regulatory interest. Deep percolation rate and, thus, net infiltration, also directly affects the transport of radionuclides from the repository horizon to the SZ.

Net infiltration is directly related to climatic and surface conditions. Precipitation occurs episodically at Yucca Mountain, with years between rain events that lead to infiltration below the surface. Near-surface processes such as evaporation, plant transpiration, and overland runoff reduce net infiltration to approximately 5 percent of total precipitation on an annual average basis (Figure 4.3.5-1). Net infiltration estimates are the highest along Yucca Mountain crest and the eastward trending ridge tops, because of the incidence of thin soils, precipitation increasing as a function of elevation, intermediate permeability of the caprock units, and high permeability of the open and soil-filled fractures. Surface water runs off toward channels and the toes of steep slopes and can increase net infiltration at these locations, though these locations tend to have relatively small surface areas.

Thin soil layers allow infiltration to enter fractures in the underlying bedrock more quickly and, thus, escape loss through evaporation. Simulations of bare soil infiltration indicate that mean annual infiltration is strongly dependent on surface soil thickness (Stothoff, et al., 1997; Stothoff, 1999). Mean annual infiltration estimates are generally higher for areas where soil thickness is less than 0.5 m (20 in). Whereas exposed bedrock promotes runoff, the high permeability of soils allows for storage of water from intense rain events. Once the water capacity of thin soils is filled, open and filled fractures in the bedrock readily transmit water to sufficient depth beyond the reach of transpiring plant roots, thus becoming net infiltration.

At Yucca Mountain, most of the repository footprint is overlain by thin soil layers less than 0.5-m (20-in) thick, with significant variability across the site. Both the DOE and NRC soil depth models qualitatively represent the system, in their respective performance assessment models, with consideration of the significant variability across the site.

The spatial variation in precipitation, soil thickness, and bedrock properties over the repository footprint have been explicitly incorporated into the calculation of mean annual infiltration, and thus deep percolation, for each subarea of the repository.

Using TPA Version 4.1 in sensitivity analyses, Mohanty, et al. (2004) determined that the mean areal average infiltration into the subsurface was one of the two most influential parameters corresponding to overall peak risk (Table 4.3.4-1). The peak dose estimates from each realization were also found to be most sensitive to the mean areal average infiltration into the subsurface (Figure 4.3.5-2). In addition, the subarea wetted fraction, which is correlated to mean annual net infiltration, was also found to be an influential parameter (Table 4.3.4-1 and Figure 4.3.5-2).

<u>Uncertainties</u>

Estimates of uncertainty for net infiltration are mostly based on modeling analyses that take into consideration the uncertainty in individual model parameters. The average net infiltration rate for the repository footprint area is estimated, in the TPA Version 4.1 code, to range from 4 to 13 mm/yr (0.2 to 0.5 in/yr) for the modern climate. The DOE performance assessments probabilistically consider a set of low-, medium-, and high-infiltration scenarios, which, for the modern climate conditions, spans a similar range of area-averaged infiltration rates. Reducing the uncertainties in net-infiltration estimates remains difficult because it is not possible to accurately measure net infiltration in environments where thin soils overlay bedrock. It is also difficult to estimate appropriate representative values for bulk bedrock permeability and soil thickness, which are important input parameters for net-infiltration models. Nevertheless, confidence in the current range of uncertainty estimated for net-infiltration rates is gained through other lines of evidence, such as analysis of temperature data and chloride content in perched water and in rock matrix.

A representative soil thickness is not easily supported by measurements because of the high degree of natural variability over small spatial scales. However, because most of the repository footprint at Yucca Mountain is estimated to be overlain by soil layers less than 0.5 m (20 in) thick, and is correspondingly represented in the DOE and NRC soil depth models, the uncertainty in the soil cover thickness would not likely lead to overly optimistic estimates of repository performance.

Long-term Climatic Change: Medium Significance to Waste Isolation

Long-term climatic change, in terms of changes in precipitation and temperature, will directly affect the rate of net infiltration and, subsequently, deep percolation rate.

<u>Discussion</u>

One of the main processes that control net infiltration is climatic conditions, expressed in terms of temporal and spatial variation of precipitation and temperature. Annual net infiltration is expected to vary over the long term (i.e., thousands of years) because of variation in temperature and precipitation. Based on historical data and paleoclimatic markers, a full cycle of climatic changes is assumed to occur roughly every 100,000 years. Monsoonal conditions (wetter and hotter than present) and glacial transition conditions (wetter and cooler) may occur within the next 10,000 years.

In the NRC's analyses (e.g., the TPA Version 4.1 base-case scenario), the full glacial climate is assumed not to occur during the 10,000-yr performance period. Figure 4.3.5-3 shows the mean net infiltration rates across all subareas is expected to increase from 8 mm/yr (0.3 in/yr) for the present-day climate to 15 mm/yr (0.6 in/yr) during the 10,000-year period of regulatory interest (i.e., a partial transition to a glacial climate), which is less than a factor of two increase. During a full glacial period starting at 30,000 years, the net infiltration rate is expected to range between a minimum of 4 mm/yr (0.2 in/yr) to a maximum of 30 mm/yr (1.2 in/yr).

In particular, the DOE performance assessment approach assumes an early and instantaneous transition to a monsoonal climate in an average time of 600 years from present and another instantaneous change to a glacial transition climate in about 2000 years from present. These climate changes have a significant effect on net infiltration estimates. For example, in the DOE medium-infiltration case, the area-averaged net infiltration over the UZ model domain is increased from 4.6 to 12.2 mm/yr (0.2 to 0.5 in/yr), after the change to a monsoonal climate, and subsequently increases to 17.8 mm/yr (0.7 in/yr) for the glacial transition climate. Thus, the DOE performance assessment approach considers approximately a factor-of-4 increase in net infiltration (for the medium-infiltration case) as a result of climate change during the performance period. As previously discussed in this section, this increased net infiltration significantly affects both drift seepage rates and the rate of radionuclide transport in the UZ.

<u>Uncertainties</u>

Important uncertainties pertaining to climate change are the timing of the onset of climate change, and the magnitude of temperature and precipitation changes that may occur as a result of the climate changes. The NRC and DOE use different approaches to estimate future climatic conditions. The NRC uses a smooth transition from the modern climate to a glacial-transition climate, combined with random sampling of a precipitation multiplier and a temperature shift. The DOE uses an instantaneous step-function approach, combined with upper-bound, mean, and lower-bound precipitation and temperature records, which results in higher estimates of net infiltration rates over the 10,000-year performance period. The DOE step-function approach is based on recent evidence presented in the scientific community supporting much faster climate transitions than previously believed likely to occur. The DOE upper-bound, mean, and lower-bound precipitation and temperature records are based on measurements obtained from a range of analog sites that are believed to adequately bound the likely magnitude of climate changes that might occur at Yucca Mountain.

Neither the NRC nor DOE approaches consider the potential effects of anthropogenically induced global warming. In general, the exclusion of anthropogenic effects on climate is believed to be conservative because global warming would increase temperature and reduce precipitation in the Yucca Mountain region.

Another model uncertainty is that periods of climate transition may lead to increased net infiltration as vegetation and soil thickness (e.g., erosion) do not immediately adjust to the new climate conditions; however, climate-induced changes in vegetation and soil cover would be relatively short-lived compared to the 10,000 year performance period, and thus this uncertainty is believed to be relatively unimportant.

4.3.5.2 References

Mohanty, S., et al. "System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code." CNWRA 2002-05. Revision 2. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2004.

Stothoff, S.A., H.M. Castellaw, and A.C. Bagtzoglou. "Simulating the Spatial Distribution of Infiltration at Yucca Mountain, Nevada." San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 1997.

Stothoff, S. "Infiltration Abstractions for Shallow Soil over Fractured Bedrock in a Semiarid Climate." San Antonio, Texas: Center for Nuclear Regulatory Analyses. 1999.

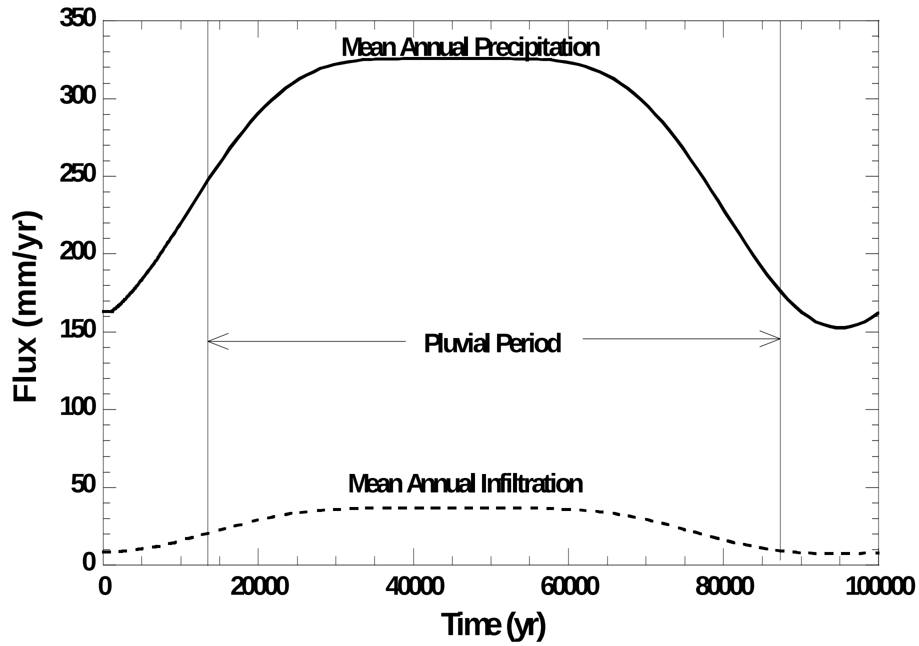
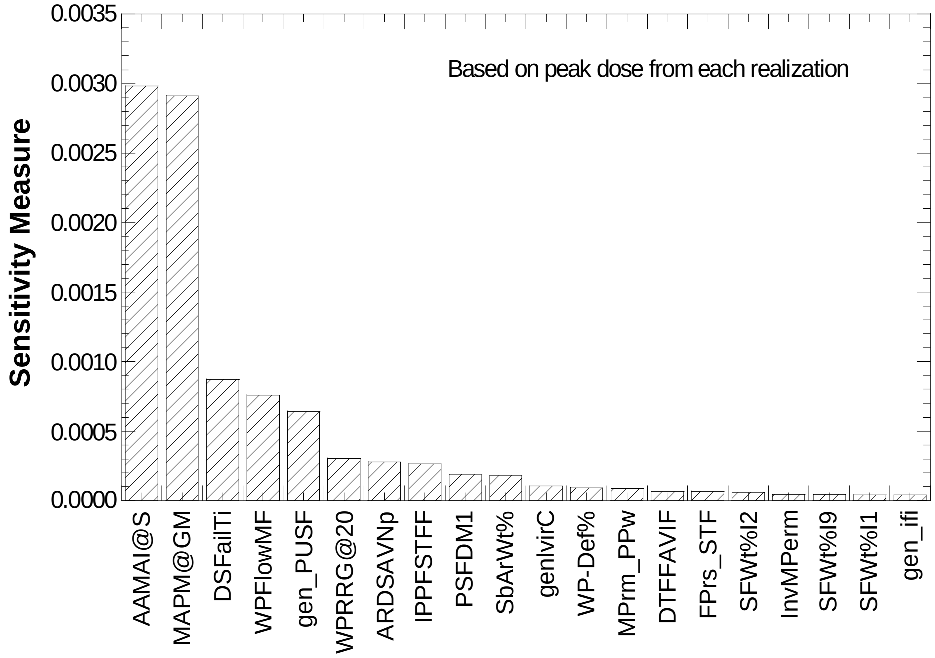


Figure 4.3.5-1. Mean annual precipitation and infiltration at the repository horizon averaged over all subareas and encompassing both the current and pluvial periods for the mean value data set. (From Mohanty, et al., 2004, Figure 3-1)



Note:

AAMAI@S	Areal average mean annual infiltration at start
MAPM@GM	Mean annual precipitation increase at glacial maximum
DSFailTi	Time of failure of the drip shield
WPFlowMF	Waste package flow multiplication factor
gen_PUSF	Plant uptake scaling factor
WPRRG@20	Well pumping rate within 20 km
ARDSAVNp	Alluvium matrix retardation for neptunium
IPPFSTFF	Effective fraction of saturated rock matrix accessible to matrix diffusion
PSFDM1	Preexponential factor for spent fuel dissolution rate
SbArWt%	Subarea wet fraction
genlvirC	Current leafy vegetable irrigation rate
WP-Def%	Fraction of total waste packages in a subarea that fails at time t
MPrm_PPw	Prow Pass-welded matrix permeability
DTFFAVIF	Distance traveled in tuff
FPrs_STF	Fracture porosity of saturated tuff
SFWt%I2	Spent fuel wet fraction for initial failures in subarea 2
InvMPerm	Matrix permeability of the invert
SFWt%I9	Spent fuel wet fraction for initial failures in subarea 9
SFWt%I1	Spent fuel wet fraction for initial failures in subarea 1
gen_ifi	Irrigation interception fraction

Figure 4.3.5-2. Influential parameters identified using the peak dose from each realization. (From Mohanty, et al., 2004, Figure 4-6)

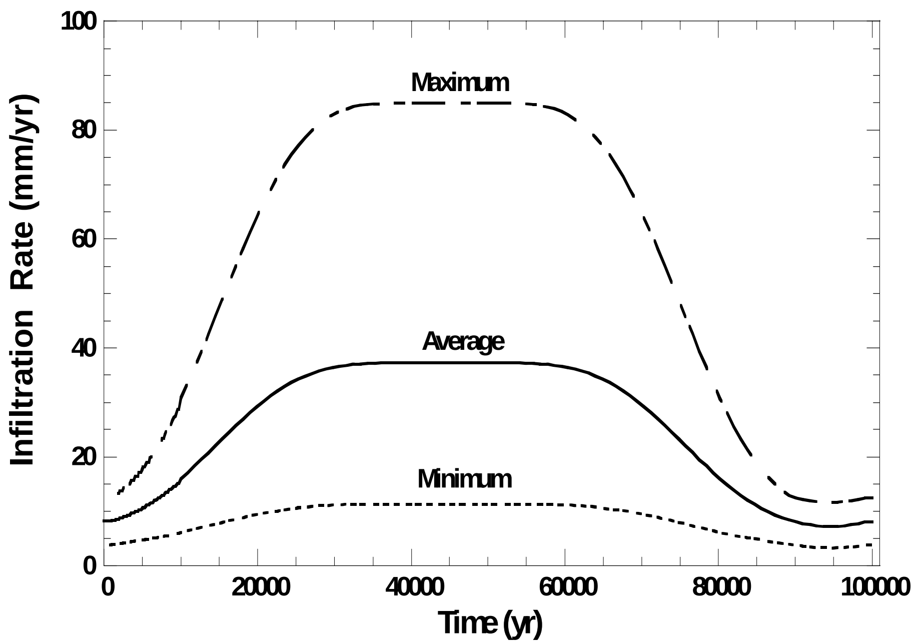


Figure 4.3.5-3. Mean, maximum, and minimum infiltration rates in the unsaturated zone for all subareas. (The subarea average infiltration rate is obtained by averaging over all 350 realizations.)
(From Mohanty, et al., 2004, Figure 3-21)

4.3.6 Flow Paths in the Unsaturated Zone (UZ2)

Risk Insights:	
Seepage	High Significance
Hydrologic Properties of the UZ	Medium Significance
Transient Percolation	Low Significance

4.3.6.1 Discussion of the Risk Insights

Seepage: High Significance to Waste Isolation

Seepage of water into the drifts determines the amount of water that comes into contact with the drip shields and waste packages and affects the release and transport of lower-solubility radionuclides (e.g., Np-237). The amount of seepage is expected to affect the formation of salts on the surfaces of the drip shield and waste packages.

<u>Discussion</u>

Seepage is important for estimating releases from the repository because: (1) the amount of seepage affects the release of radionuclides that have low solubility limits; (2) the spatial variability in seepage affects the number of waste packages that are estimated to be dripped on; and (3) the amount of seepage affects the formation salts on the surfaces of the drip shield and waste package. In part because of the technical difficulties in determining the location and extent of dripping, performance assessments have generally assumed that dripping will occur in a significant portion of the repository (e.g., on the order of 25 percent) and a portion of this dripping water will enter breached waste packages.

Releases are estimated to be greatest where dripping water (seepage) enters a waste package and radionuclides are transported by the water out of the package. Thus, the spatial variability of seepage will affect releases by affecting the number of waste packages that are dripped on. If only a small number of waste packages experience dripping, the estimates of dose would be expected to diminish accordingly.

Seepage is expected to be the primary mechanism for transporting radionuclides out of the waste package. The amount of dripping water is not expected to significantly affect radionuclides with high solubility limits (e.g., I-129, Tc-99) because estimates indicate only a small amount of water is needed to mobilize these radionuclides. However, solubility-limited radionuclides (e.g., Np-237, Am-241), which comprise the bulk of the radionuclide inventory, can be affected by the amount of seepage (Figure 4.3.6-1). However, estimates indicate these radionuclides are generally slowed or retarded during transport in the geosphere during the first 10,000 years.

Seepage also affects the rate of corrosion of the drip shield and waste package. The formation of aggressive salts due to evaporation of seepage on the waste package may result in accelerated corrosion. These issues are discussed in greater detail in Sections 4.3.1 and 4.3.3.

Uncertainties

Quantitative assessments of potential seepage of water into repository drifts and onto waste packages are complicated by factors such as heterogeneity in the UZ, thermal perturbations to the flow field, capillary processes in fracture networks intersecting large openings, drift degradation, and thermal effects. Current approaches estimate the location and amount of dripping based on a variety of information (e.g., mining experience, numerical modeling, and field experiments) to provide a range for expected behavior. It is difficult to accurately quantify the effect of the waste heat on the unsaturated flow field. However, the uncertainties cannot increase dripping by more than a factor of four, at most, because performance assessments have generally assumed dripping will occur in a significant portion of the repository (e.g., on the order of 25 percent). The waste heat is expected to cause areas of evaporation and condensation in the UZ, especially during the first few hundred years when it is greatest (e.g., waste-package temperature of approximately 170°C). The effects of this temperature change, on drift seepage and relative humidity, are important for estimating the effects of seepage water chemistry and deliquescent salt formation on corrosion rates of the drip shields or waste packages. (See also Section 4.3.3.) Generally, these effects are evaluated in performance assessments by considering different water chemistries and deliquescent salt formations in conjunction with corrosion, rather than explicitly representing the spatial and temporal variation in seepage and water chemistries.

Hydrologic Properties of the Unsaturated Zone: Medium Significance to Waste Isolation

For UZ flow paths that occur mainly within fractured welded or zeolitized tuff units, where matrix conductivities can be significantly lower than the percolation rate, the unretarded radionuclide travel times from the repository horizon to the water table are on the order of a few tens of years because water flows primarily in fractures. Longer UZ travel times, on the order of several hundreds of years, are estimated for areas beneath the repository where the CHnv unit is present. The longer travel times in the CHnv unit are attributed to its relatively large matrix conductivity such that water tends to flow in the matrix rather than the fractures. The areal extent and thickness of the CHnv unit are considered to be moderately important aspects of UZ flow and transport.

Discussion

The CHnv unit is characterized by a relatively large matrix hydraulic conductivity; thus, for the range of anticipated percolation rates, the water flow in this unit is expected to remain in the matrix (i.e., no fracture flow) and be slow. Currently, the NRC's performance assessment estimates that approximately half of the repository footprint will be underlain by sufficient thickness of the CHnv unit to have a significant effect on performance. If the thickness were to increase over a significantly larger portion of the footprint (e.g., 90 percent or greater) the UZ below the repository would have an ever increasing effect on performance. For example, Winterle, et al. (1999, Figure 2-2) evaluated the effects of CHnv extent and thickness using the TPA Version 3.2 code. The original base-case model for the TPA Version 3.2 code represented the CHnv unit beneath only two of seven repository subareas. Winterle, et al. (1999) stated that available borehole data suggest the presence of at least thin lenses of nonwelded vitric layers [as thin as 2 m (6.6 ft)] underneath all repository subareas. The Winterle, et al. (1999) analysis with the TPA Version 3.2 code indicated that consideration of

CHnv layers under all subareas resulted in a reduction in the 10,000-year peak dose by a factor of about four.

Additionally, figure 4.3.6-2 depicts unretarded radionuclide travel times from TPA 4.1 analyses that fall into two distinct categories. The first category reflects flow paths with rather short UZ travel times (tens of years) and is representative of portions of the repository where the CHnv unit is not present beneath the repository (Figure 4.3.6-3). The second category represents much longer travel times (on the order of several hundreds of years) and is representative of areas beneath the repository primarily where the CHnv unit is present (Mohanty, et al., 2004, Section 3.3.5). If the unretarded travel time is 500 years or longer, and the retardation factor is at least 20, radionuclides will not reach the water table within the regulatory period of 10,000 years. Current information indicates that a significant fraction of the inventory of the repository has retardation factors greater than 20. (See Section 4.3.7.) Current information indicates that approximately half the repository footprint is underlain by a sufficient thickness of the CHnv unit to have a significant effect on UZ unretarded radionuclide travel times. The DOE TSPA includes a similar areal and thickness distribution of the CHnv unit, as is modeled in TPA. The DOE sensitivity studies of UZ transport times, presented at the radionuclide transport technical exchange held in December, 2000, also indicate that the CHnv unit is important for longer travel times.

Uncertainties

The thickness and areal extent of the CHnv directly below the repository are difficult to estimate precisely because there is a limited number of exploratory boreholes within the proposed repository footprint. However, previous analyses have assessed the affect of this uncertainty (Winterle et al., 1999).

The flow paths in the UZ are also uncertain because of the potential for focusing of flow into "fingers" of flow rather than uniform flow or horizontal diversion of flow. It is anticipated that perturbations from focused flow will not have a significant potential for increasing consequences. For example, horizontal diversion of flow around low conductivity increases the flow path in a higher-conductivity zone, where there will be greater potential for matrix flow.

Values for the matrix conductivity of the UZ tuff units are important when the matrix conductivity is sufficiently large that a significant fraction of the percolation can be expected to flow in the matrix (e.g., conductivities at least 50 percent of the percolation rates). Representation of the uncertainty and variability in matrix conductivity in the range of the anticipated percolation rates [e.g., 10 mm/yr (0.4 in/yr)] is an important aspect of the UZ performance.

Transient Percolation: Low Significance to Waste Isolation

Episodic or transient percolation through the UZ below the root zone caused by short-term variation in precipitation does not significantly affect the spatial nor temporal variability of seepage into the drifts. After a precipitation event, infiltrating water moves in pulses vertically through the fractured rock unit and into the underlying rock units, where the pulses are variably damped in the Paintbrush Tuff non-welded unit into more steady vertical flow.

Discussion

Rainfall at the Yucca Mountain site is highly episodic, generally occurring over short periods of time. A small amount of the annual rainfall is estimated to contribute to net infiltration, whereas the bulk of the rainfall at Yucca Mountain is lost to evapotranspiration and runoff. Current estimates suggest, on average, approximately 5 percent of the rainfall contributes to infiltration (Figure 4.3.6-4). Although the infiltration near the surface is expected to be episodic, the pulses of infiltration are expected to be dampened as the infiltration moves deep (e.g., many tens of meters) below the surface. Although the rate may change from year to year, it is modeled as constant within any given year. As shown in Figure 4.3.6-4, the net infiltration (assumed to be the same as seepage in this figure) does not vary significantly despite larger changes in the precipitation rate. This is especially true for the initial 10,000 years, because infiltration at later years is influenced by cooler temperatures associated with a glacial period.

Uncertainties

Estimates of future short-, intermediate-, and long-term variations in precipitation are uncertain. Both the NRC and DOE performance assessment models currently account for effects of short-term weather patterns and long-term climate changes. The effect of short-term variations are incorporated using 10-yr records of hourly meteorological data, in process-level models, to support the TPA abstraction. Although the episodic nature of rainfall has an effect on the net infiltration, transient percolation is not expected to have a significant effect on seepage in the repository drifts, because of damping of flow variations in geologic units above the repository. The effects of long-term climate variations are addressed by including precipitation and temperature shifts, based on historical information on climate change, in performance assessments.

4.3.6.2 References

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Mohanty, S., et al. "System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code." CNWRA 2002-05. Revision 2. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2004.

Winterle, J.R., R.W. Fedors, D.L. Hughson, and S. Stothoff. “Update of Hydrologic Parameters for the Total-System Performance Assessment Code.” Letter report. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. August 1999.

Water Flow into Waste Package

Water Influx Sensitivity

[packages needed to release 15 mrem/yr at drift wall - no geologic delay]

Nuclide	WP breach at 5,000 yrs		WP breach at 1,000 yrs	
	Low Flow	High Flow	Low Flow	High Flow
Tc 99	>7,000	>7,000	>7,000	3,100
I 129	>7,000	>7,000	>7,000	6,700
Np 237	>7,000	120	>7,000	40
Am 241	>7,000	65	>7,000	1
Pu 240	>7,000	2	>7,000	1

Figure 4.3.6-1. Sensitivity to water influx.
[waste package (WP); technetium-99 (Tc-99); iodine-129 (I-129); neptunium-237 (Np-237); americium-241 (Am-241); plutonium-240 (Pu-240)]
(From McCartin, 2003)

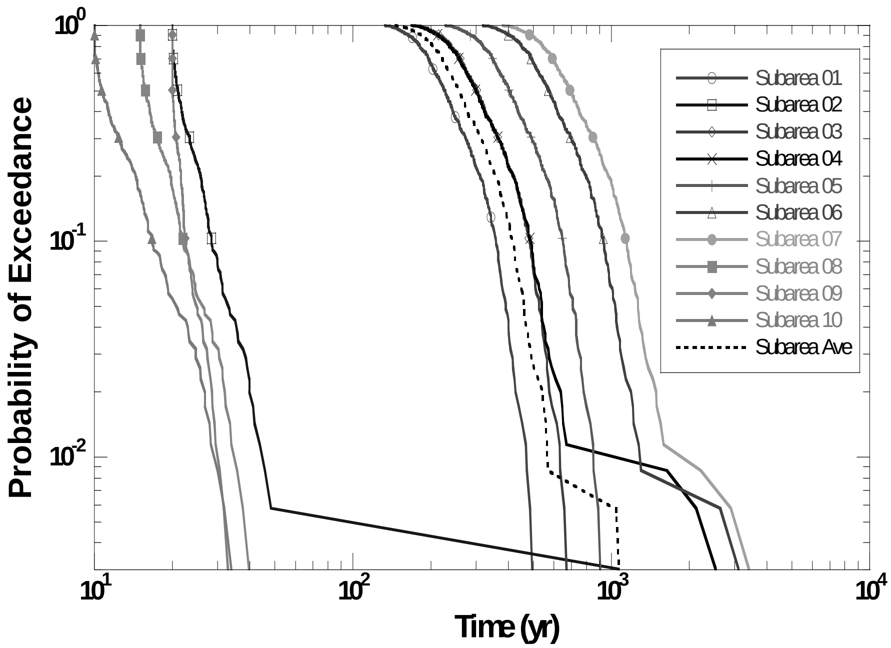

Figure 4.3.6-2. Complementary cumulative distribution function of unsaturated zone unretarded radionuclide travel times for each of the 10 repository subareas and the average of all 10 areas (based on 350 realizations). (From Mohanty, et al., 2004, Figure 3-30)

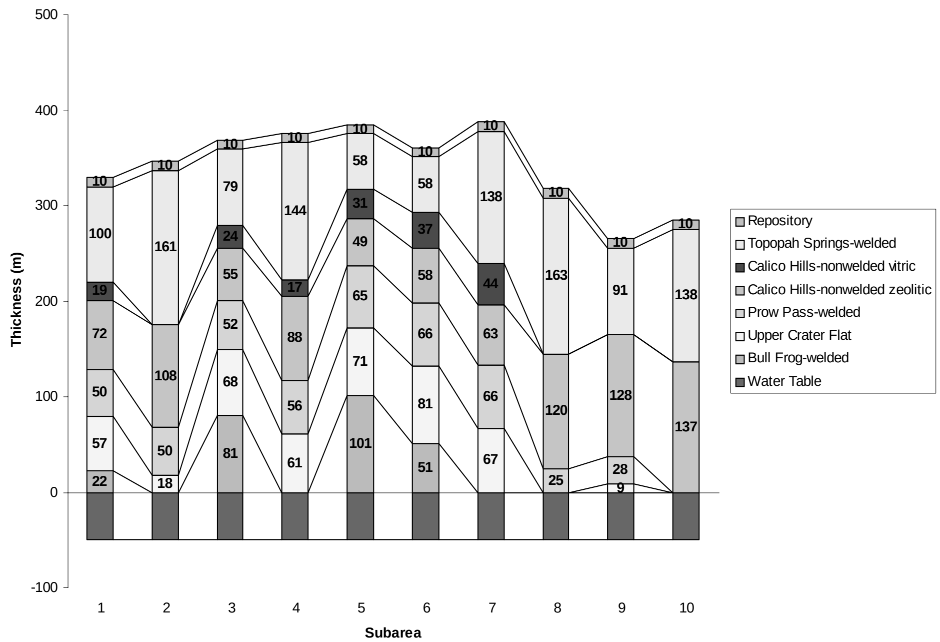

Figure 4.3.6-3. Depiction of the stratigraphic thicknesses below each of the 10 repository subareas. (From Mohanty, et al., 2004, Figure 3-9)

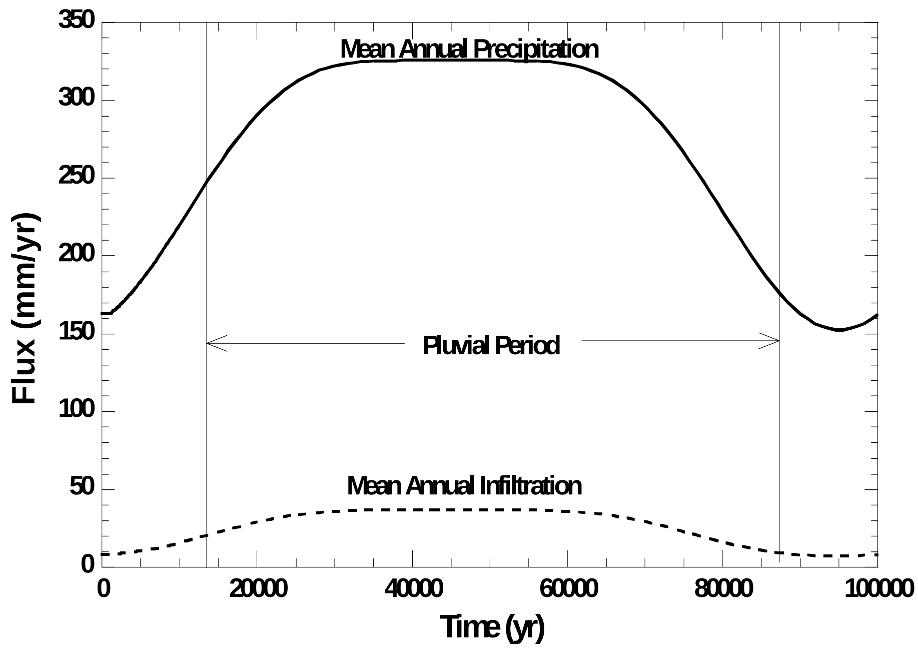

Figure 4.3.6-4. Predicted mean annual precipitation and infiltration at the repository horizon averaged over all subareas and encompassing both the current and pluvial periods for the mean value data set. (From Mohanty, et al., 2004, Figure 3-1)

4.3.7 Radionuclide Transport in the Unsaturated Zone (UZ3)

Risk Insights:	
Retardation in the Calico Hills non-welded vitric Unit	Medium Significance
Matrix Diffusion in the Usaturated Zone	Medium Significance
Effect of Colloids on Transport in the Usaturated Zone	Medium Significance

4.3.7.1 Discussion of the Risk Insights

Retardation in the Calico Hills Non-welded Vitric Unit: Medium Significance to Waste Isolation

Retardation in the CHnv unit has the potential to delay the movement of most radionuclides for very long time periods (e.g., thousands to tens of thousands of years and longer) for nuclides that tend to sorb onto rock surfaces (e.g., Np-237, Am-241, and Pu-240). Certain nuclides do not readily sorb onto rock surfaces (i.e., I-129 and Tc-99). Where the CHnv unit is present below the repository, sorption of radionuclides may limit releases to the SZ, within the compliance period, to insignificant quantities for all radionuclides except I-129 and Tc-99. In this context the retardation factor for Np-237 is the most significant because of the large inventory and long half-life for this radionuclide.

Discussion

Retardation in the CHnv has the potential to delay the transport of sorbing radionuclides for time periods on the order of 10,000 years and beyond. Figure 4.3.7-1 shows the effect of the range of retardation in the CHnv unit on the transport time for key radionuclides through the UZ to the water table (i.e., SZ). Radionuclides that do not readily sorb onto rock surfaces (i.e., I-129 and Tc-99) show limited delay time (i.e., 450 years) relative to sorbing or retarded radionuclides (i.e., Np-237, Am-241, Pu-241) where releases are delayed on the order of 10,000 years for low retardation factors and significantly greater for high retardation factors. It is important to note that non-sorbing radionuclides such as I-129 and Tc-99 represent a small fraction (less than 1 percent) of the overall inventory of the repository, whereas sorbing radionuclides, such as Np-237, Pu-240, and Am-241 represent a large fraction (greater than 99 percent) of the inventory (Table 4.1-1) (McCartin, 2003).

Uncertainties

The proclivity for matrix flow in the high-conductivity CHnv is key to the significant retardation provided by this unit. As described in Section 4.3.6, the areal extent and thickness of the CHnv between the repository and the water table are important uncertainties.

Matrix Diffusion in the Unsaturated Zone: Medium Significance to Waste Isolation

Matrix diffusion may have an effect on delaying radionuclide transport in the unsaturated units where the water flow is primarily in fractures.

Discussion

Radionuclides transported within fractures may be delayed because of diffusion from the fracture water into matrix water (i.e., matrix diffusion) when radionuclide concentrations are higher within the fracture water versus the matrix water. This process will affect all radionuclides. However, radionuclides that sorb onto rock surfaces (i.e., are retarded) will show longer delays than those radionuclides that are not sorbed. The NRC's modeling approach in TPA Version 4.1 for UZ flow and transport estimates matrix diffusion is likely minor in fractured tuffs in the UZ because the estimated effect in the SZ is also limited (Winterle et al., 1999, Figure 2-3). However, the DOE sensitivity analyses for matrix diffusion in the UZ indicate that a significant reduction in the simulated dose-rate history occurs when credit is taken for matrix diffusion (CRWMS M&O, 2000, Figure 5.2-14). Figure 4.3.7-2 shows the reduction is highly time-dependent and ranges from a factor of 2 to more than a factor of 10. Another example of the effects of including matrix diffusion in the UZ transport model is provided by the impact of the Topopah Spring welded tuff (TSw) unit (flow primarily in fractures) on the short travel times for ground water through the UZ. The DOE sensitivity studies indicate that travel times are reduced by an order of magnitude when matrix diffusion is not included for the TSw unit (Eddebbarh, A., et al., 2000).

Uncertainties

The process of matrix diffusion is uncertain because of complexities of the interaction of potentially fast-moving water (e.g., meters per year) in fractures with the matrix water. Differences in the chemistries of fracture water and matrix water suggest that the interactions between fracture and matrix water may not be very significant (Murphy and Pabalan, 1994).

Effect of Colloids on Transport in the Unsaturated Zone: Medium Significance to Waste Isolation

Transport of radionuclides attached to natural colloids may reduce the effectiveness of sorption properties of the CHnv unit.

Discussion

Radionuclides that attach to colloids have the potential to be transported in a manner that may substantially reduce or eliminate the beneficial effect of sorption in geologic materials such as the CHnv unit. Although performance effects have not been explicitly examined using the TPA code, the code was used to bound colloid effects by allowing unretarded transport of relatively immobile actinides such as plutonium, americium, and thorium. That conservative analysis, which assumes a bounding unretarded transport, yielded an increase in dose by more than a factor of 10 (Figure 4.3.4-6). The DOE has performed analyses to evaluate the sensitivity of colloids where plutonium and americium are irreversibly sorbed to waste form colloids (Figure 4.3.4-7). The DOE analyses indicated that colloidal concentrations were only significant in 10,000 years, under an intrusive igneous event, wherein a large number of waste packages are significantly damaged. The potential importance of colloidal transport, if all waste packages were to fail within 10,000 years, is evident in the DOE TSPA-SR (CRWMS M&O, 2000), which indicated colloidal plutonium is the second highest dose contributor after 70,000 years, when a significant number of waste packages are estimated to have failed, allowing release of radionuclides.

Uncertainties

Some field studies (Kersting, et al., 1999) suggest that colloids can travel relatively easily under particular conditions, which are not expected to be relevant to transport in the UZ at Yucca Mountain. There is considerable uncertainty both with the determination of the colloidal concentration and the extent to which colloids will be transported or filtered in the geosphere as they move through small fracture openings and matrix pores.

4.3.7.2 References

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR-WIS-PA-000001 REV 00 ICN 01. Las Vegas, Nevada: Civilian Radioactive Waste Management System Management and Operating Contractor. 2000.

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Unsaturated Zone

Retardation Sensitivity (CHnv)

[years for initial release into Unsat zone to exit Unsat zone]

Nuclide	Rf (low)	Rf (high)
Tc 99	450	450
I 129	450	450
Np 237	9,000	60,000
Am 241	>100K	>100K
Pu 240	>100K	>100K

Figure 4.3.7-1. Unsaturated zone retardation (Rf) sensitivity in the Calico Hills non-welded vitric unit (CHnv). [technetium-99 (Tc-99); iodine-129 (I-129); neptunium-237 (Np-237); americium-241 (Am-241); plutonium-240 (Pu-240); thousand (K)] (From McCartin, 2003)

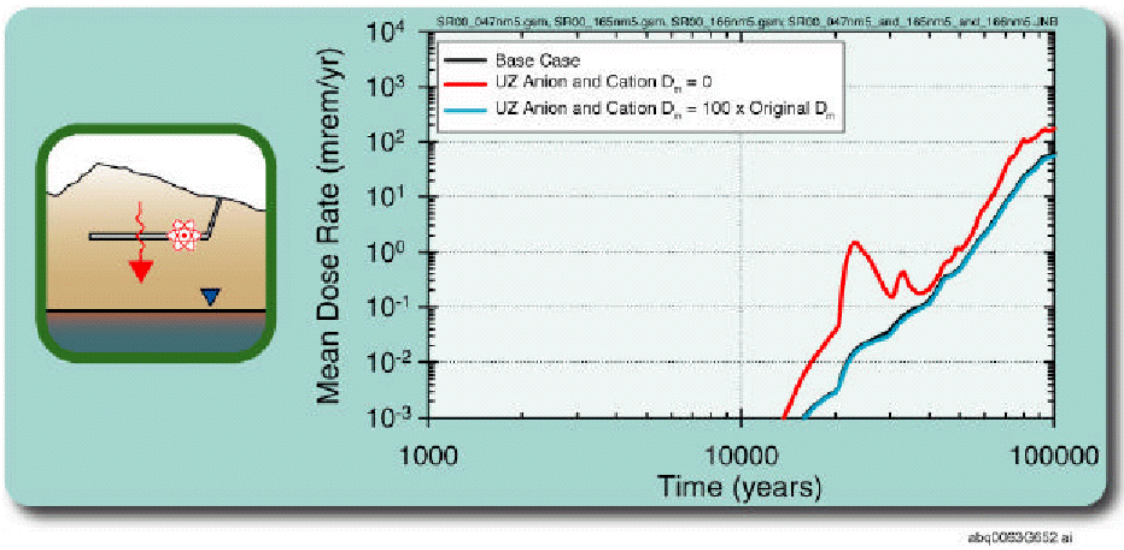

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Figure 4.3.7-2. Sensitivity to matrix diffusion in the unsaturated zone. (From CRWMS M&O, 2000; page F5-37)

4.3.8 Flow Paths in the Saturated Zone (SZ1)

Risk Insights:	
Saturated Alluvium Transport Distance	Medium Significance

4.3.8.1 Discussion of the Risk Insights

Saturated Alluvium Transport Distance: Medium Significance to Waste Isolation

The saturated flow path is comprised of both fractured tufaceous rock and porous alluvium. Alluvium comprising a portion of the flow path is important because of the large capacity of the alluvium to retard a majority of the radionuclides. To have a significant influence on retarded radionuclides, the alluvium needs to comprise at least 500 m (1640 ft) of the total flow path of 18 km (11.2 mi).

<u>Discussion</u>

Both fractured tufaceous rock and porous alluvium comprise the saturated flow path. The velocity of the water flow within these two units can be quite different because of differences in the hydrologic properties, specifically between the porosity of the alluvium (i.e., on the order of 15 percent of the overall volume) and the porosity of the fractured tuff (i.e., on the order of 0.1 to 1 percent of the overall volume). The unretarded radionuclide travel time for the SZ is estimated to be on the order of several hundreds of years and longer (Figure 4.3.8-1). Because flow velocities in the alluvium are small relative to the fractured tuff, the majority of the travel time is in the alluvium. Radionuclides traveling through the alluvium are especially important because of the potential capability of the porous media to delay a majority of radionuclides due to chemical sorption onto mineral surfaces. (See Section 4.3.9.) An alluvium flow path length of only 500 m (1640 ft) [relative to a total SZ flow path length of 18 km (11.2 mi)] has a significant capacity to retard radionuclides (Figure 4.3.8-2). Current information indicates alluvium is present for at least 2 km (1.2 mi) along the flow path.

<u>Uncertainties</u>

Based on the results shown in Figure 4.3.8-2, uncertainty in the length of the flow path in the alluvium [i.e., 1 km (0.6 mi) versus 5 km (3.1 mi)] does not have a significant impact on performance, and the variation of retardation factor is significant only for Np-237. Flow within the SZ is affected by heterogeneity. For example, variations in structure (e.g., fault zones) and permeability (e.g., high-permeability zones) are present. Although these types of heterogeneity are expected to result in local perturbations in the flow field, the flow regime, on a regional scale, is not expected to be significantly altered. Detailed hydrologic modeling studies of the SZ that have examined the effects of fault zones on flow and transport (Figure 4.3.8-3) indicate inclusion of additional structure in the model would affect the spreading of a contaminant plume, but would not significantly affect the unretarded radionuclide travel time. For example, the presence of a fault tends to spread pathlines vertically (Figure 4.3.8-3a) while retaining the general leading-edge shape of the plume (Figure 4.3.8-3b).

Borehole data suggest that alluvial sediments can be strongly heterogeneous, ranging from fine-grained clay sediments to coarse gravels and sands. The effect of this heterogeneity on flow paths and travel times in saturated alluvium is an important uncertainty that is handled in both the DOE and NRC performance assessment models by stochastically sampling the effective porosity of alluvium. Lower values of effective porosity have the effect of increasing ground water velocity and thus decreasing unretarded radionuclide travel time estimates.

4.3.8.2 References

McCartin, T.J. "Understanding Performance Assessment Results." Presentation to the Advisory Committee on Nuclear Waste. Washington, DC: U.S. Nuclear Regulatory Commission. March 2003.

Mohanty, S., et al. "System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code." CNWRA 2002-05. Revision 2. San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2004.

NRC. NUREG-1538, "Preliminary Performance-Based Analyses Relevant to Dose-Based Performance Measures for a Proposed Geologic Repository at Yucca Mountain." Washington, DC: U.S. Nuclear Regulatory Commission. 2001.

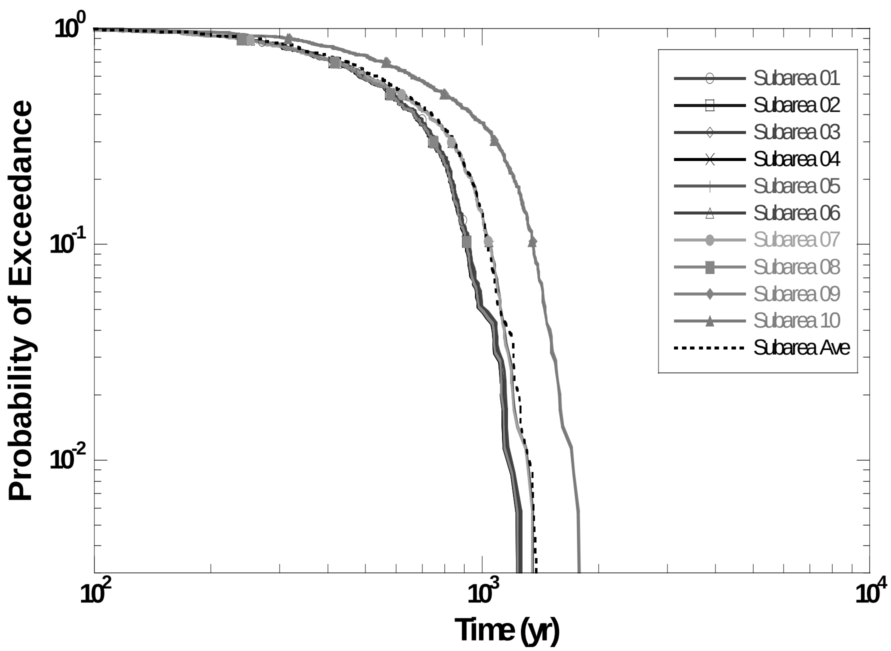

Figure 4.3.8-1. Complementary cumulative distribution function of saturated zone unretarded radionuclide travel times for 350 realizations. (From Mohanty, et al., 2004, Figure 3-33)

Saturated Zone

Retardation Sensitivity

[years for initial release into Sat zone to exit Sat zone]

Nuclide	Alluv(1km) Rf (low)	Alluv(1km) Rf (high)	Alluv (5km) Rf (low)	Alluv(5km) Rf (high)
Tc 99	350	350	550	550
I 129	350	350	550	550
Np 237	950	76,000	1,050	>100K
Am 241	>100K	>100K	>100K	>100K
Pu 240	54,000	>100K	>100K	>100K

Figure 4.3.8-2. Saturated zone retardation (Rf) sensitivity. [technetium-99 (Tc-99); iodine-129 (I-129); neptunium-237 (Np-237); americium-241 (Am-241); plutonium-240 (Pu-240); thousand (K)] (From McCartin, 2003)

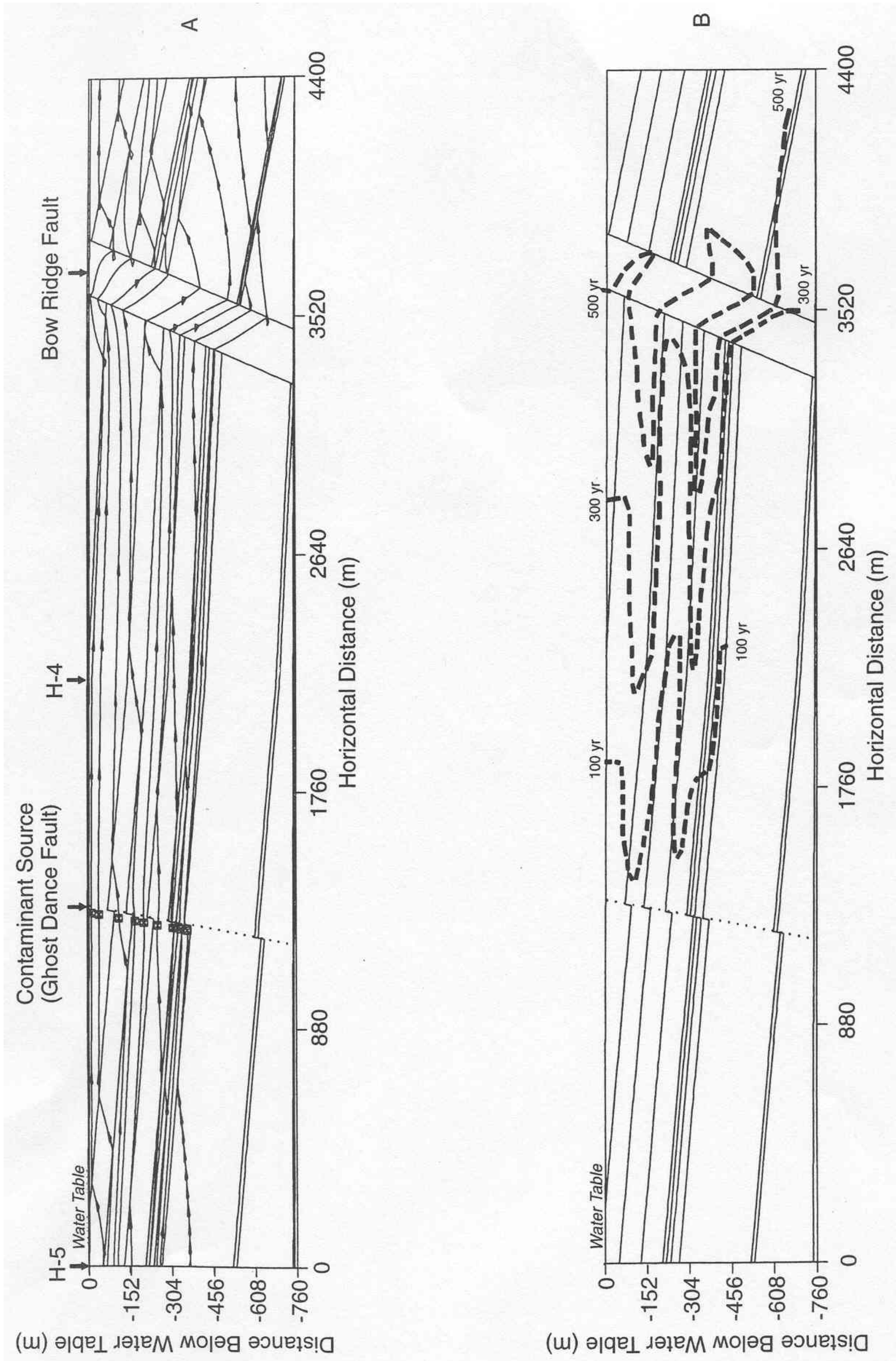

Figure 4.3.8-3. Effect of faults on pathlines (a) and unretarded travel times (b) for vertical cross-section flow model. (From NRC, 2001, Figure 3-9)

4.3.9 Radionuclide Transport in the Saturated Zone (SZ2)

Risk Insights:	
Retardation in the Saturated Alluvium	High Significance
Matrix Diffusion in the Saturated Zone	Medium Significance
Effect of Colloids on Transport in the Saturated Zone	Medium Significance

4.3.9.1 Discussion of the Risk Insights

Retardation in Saturated Alluvium: High Significance to Waste Isolation

Retardation in the alluvium unit has the potential to delay the movement of most radionuclides for very long time periods (e.g., thousands to tens of thousands of years and longer) for nuclides that tend to sorb onto porous materials (e.g., Np-237, Am-241, Pu-240). In this context, Np-237 is the most significant radionuclide affected by retardation in the alluvium, because of the large inventory and long half-life for this radionuclide.

Discussion

The transport of the vast majority of the radionuclides is expected to be significantly delayed by the saturated alluvium, because of chemical sorption on mineral surfaces (Figure 4.3.8-2). Although certain radionuclides (e.g., I-129, and Tc-99) are not typically sorbed onto mineral surfaces under the geochemical conditions that predominate in the SZ, these radionuclides comprise a small fraction of the inventory of spent nuclear fuel (i.e., less than 1 percent). In contrast, radionuclides such as Am-241 and Pu-240, which comprise a majority of the inventory of spent nuclear fuel (Figure 4.1-1), are characterized by delay times in the alluvium on the order of tens of thousands of years and greater for the full range of expected retardation factors for these nuclides. The range of expected retardation factors for Np-237 is characterized by limited delays (i.e., on the order of 1000 years) to very significant delay times (i.e., on the order of 100,000 years). Because most radionuclides are retarded or delayed in the alluvium, estimates of dose are typically characterized by the releases of only three radionuclides (I-129, Tc-99, and Np-237). Initial dose is typically from I-129 and Tc-99 and is followed by a later peak, from Np-237. The estimated dose from Np-237 tends to be larger than the estimated dose from I-129 and Tc-99 because of the large dose-conversion factor associated with Np-237.

Uncertainties

The DOE sensitivity studies also indicate that the retardation factor used for Np-237 in the saturated alluvium has a significant impact on Np-237 travel time through the alluvium. The range of the retardation factor used for Np-237 depends on the geochemistry and mineralogy of the SZ. Although the retardation factors currently used in the DOE performance assessment of Yucca Mountain likely provide a reasonable estimate of Np-237 sorption, the technical bases for these values are based on experiments with limited accounting of SZ chemistry or variation in alluvium mineralogy. The effectiveness of the alluvium in delaying certain radionuclides may be lessened if they are transported as colloids (further discussion of colloids is provided below). Current data for Np-237 transport parameters does not include colloids.

Matrix Diffusion in the Saturated Zone: Medium Significance to Waste Isolation

Matrix diffusion is a somewhat effective process for delaying radionuclides, especially those radionuclides that are sorbed onto rock surfaces (e.g., Np-237, Pu-240, and Am-241). The extent of the rock volume that is available for matrix diffusion, and each radionuclide's retardation factor, are the controlling factors.

Discussion

Radionuclides transported within the fractures of the saturated tuff may be delayed because of diffusion from the fracture water into matrix water (i.e., matrix diffusion) when radionuclide concentrations are higher within the fracture water than within the matrix water. This process will affect all radionuclides; however, radionuclides that sorb onto rock surfaces will show longer delays than those radionuclides that are not sorbed. For example, the delay time for Np-237 in the fractured tuff is increased by 1100 years, when matrix diffusion is varied from low to high effectiveness (Figure 4.3.9-1). Unlike the UZ where the flow path is relatively short (i.e., a few hundreds of meters), the SZ flow path in fractured tuff is long [i.e., at least 10 km (6.2 mi)] and thus there will be a longer period of time for matrix diffusion to occur.

The inclusion of SZ matrix diffusion appears to have only a minimal benefit for lowering dose estimates in performance assessment models. A DOE report (Ziegler, 2002), provided in response to a KTI agreement, showed comparisons of the median radionuclide transport time for "nominal case," in the TSPA-SR, to a case with essentially no matrix diffusion (diffusion coefficient reduced 10 orders of magnitude), and to a case with enhanced matrix diffusion (flow interval spacing reduced 2 orders of magnitude). The results of the comparison showed a significant increase in radionuclide transport time for the most optimistic cases of matrix diffusion, but transport times for the nominal case were not substantially greater than for the case with essentially no matrix diffusion. Winterle, et al. (1999, Figure 2-3) evaluated the effects of matrix diffusion in the SZ using the TPA Version 3.2 code. This analysis indicated that the upper bound of the diffusion parameter uncertainty distribution reduced the effective peak dose by less than 10 percent during a 10,000-year performance simulation. (The 10,000-year doses are primarily from I-129 and Tc-99, which are non-retarded radionuclides.)

Uncertainties

Uncertainties in factors that affect matrix diffusion in the SZ include the effective spacings between flowing fractures, the extent to which fracture surfaces are coated with secondary minerals, and effective *in situ* diffusion coefficients for various radionuclides. These uncertainties have led to development of performance assessment abstractions, by both the NRC and DOE, that use conservative or bounding approaches for estimating the effect of matrix diffusion on radionuclide transport.

Effects of Colloids on Transport in the Saturated Zone: Medium Significance to Waste Isolation

Transport of radionuclides attached to natural colloids may reduce the effectiveness of sorption properties of the alluvium.

Discussion

Radionuclides that attach to colloids have the potential to be transported in a manner that may substantially reduce or eliminate the beneficial effects of sorption in the alluvium. The TPA code was used to bound colloid effects by allowing unretarded transport of relatively immobile actinides such as plutonium, americium, and thorium. More realistic approaches for colloids would account for dissolved species as well as colloidally-bound species, reversible sorption onto colloids, and filtration of colloids. That conservative analysis, which assumes a bounding unretarded transport, yielded an increase in dose by more than a factor of 10 (Figure 4.3.4-6). However, accounting for more realism is expected to reduce the effects of colloids on dose. The DOE has performed analyses to evaluate the sensitivity of colloids where plutonium and americium are irreversibly sorbed to waste form colloids (Figure 4.3.4-7). The DOE analyses indicated that colloidal concentrations were only significant under an intrusive igneous event, wherein a large number of waste packages were significantly damaged. The potential importance of colloidal transport, if all waste packages were to fail within 10,000 years, is evident in the DOE TSPA-SR (CRWMS M&O, 2000), which indicated colloidal plutonium is the second highest dose contributor after 70,000 years, when a significant number of waste packages are estimated to have failed, allowing release of radionuclides.

Uncertainties

There is uncertainty both with the determination of the colloidal concentration and the extent to which colloids will be transported in the geosphere (i.e., will colloids be “filtered” as they move through small fracture openings and matrix pores). Some field studies suggest that colloids can travel relatively easily under particular conditions (Kersting, et al., 1999). Sensitivity analyses of colloid-facilitated transport models indicate that filtration is the most important parameter in understanding and estimating the importance of colloids to radionuclide transport (Cvetkovic, et al., 2002).

4.3.9.2 References

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Saturated Zone

Retardation Sensitivity *(with Mat. Diff.)*

[years for initial release into Sat zone to exit Sat zone]

Nuclide	Alluv(1km) Rf (low) Mat Diff (low)	Alluv(1km) Rf (low) Mat Dif (high)	Alluv(5km) Rf (high) Mat Dif (high)
Tc 99	300	600	700
I 129	300	600	700
Np 237	700	1,800	>100K
Am 241	>100K	>100K	>100K
Pu 240	45,000	>100K	>100K

Figure 4.3.9-1. Sensitivity of delay time in saturated alluvium (Alluv) due to retardation (Rf) and matrix diffusion (Mat Diff). [technetium-99 (Tc-99); iodine-129 (I-129); neptunium-237 (Np-237); americium-241 (Am-241); plutonium-240 (Pu-240); thousand (K)] (From McCartin, 2003)

4.3.10 Volcanic Disruption of Waste Packages (DIRECT1)

Risk Insights:	
Probability of Igneous Activity	High Significance
Number of Waste Packages Affected by Eruption	High Significance
Number of Waste Packages Damaged by Intrusion	Medium Significance

4.3.10.1 Discussion of the Risk Insights

Probability of Igneous Activity: High Significance to Waste Isolation

The risk from igneous activity is directly proportional to the probability of igneous activity. Recent aeromagnetic surveys in the Yucca Mountain region improve estimates of the number of igneous events that have occurred in the past. The number, age, and location of past igneous features are used to constrain the estimates for the probability of future events.

Discussion

Sporadically throughout the past 11 million years, basaltic volcanoes have formed in the region around the potential Yucca Mountain repository site. The probability of igneous disruption is important to risk calculations, because of the relatively low likelihood of future igneous events at the potential Yucca Mountain repository site. Analyses used to demonstrate compliance with licensing requirements must factor the likelihood of a potential disruptive event into the performance calculations, in order to determine probability-weighted dose. In addition, disruptive events with likelihoods of occurrence less than 1 in 10,000 during the 10,000-year post-closure performance period (i.e., less than 1×10^{-8} per year) do not need to be included in the total system performance calculations. Most DOE estimates for the annual probability of igneous disruption at the proposed repository site range from approximately 10^{-10} to 10^{-8} (e.g., CRWMS M&O, 2000). In contrast, alternative probability estimates generally range from on approximately 10^{-8} to 10^{-7} (e.g., NRC, 1999), to values as high as 10^{-6} using Bayesian methods (Ho, 1995; Ho and Smith, 1997). None of these models have considered current uncertainties in the number and age of past volcanic events (Hill and Stamatakos, 2002).

Figure 4.3.10-1 shows an increase in the calculated probability-weighted dose from igneous eruption by about a factor of 3, when the DOE base-case mean annual probability (1.5×10^{-8}) is replaced by its 95th percentile value (4.8×10^{-8}). This result is consistent with earlier analyses indicating direct proportionality between igneous activity probability and risk (e.g., NRC, 1999; CRWMS M&O, 2000). Thus, these current differences between the DOE models and alternative probability models may affect igneous activity risk calculations by factors of 10 to 100.

Uncertainties

Potentially significant uncertainties in the probability estimate for igneous activity currently arise primarily from uncertainties in the number, age, and composition of basaltic volcanoes possibly buried around Yucca Mountain. Variations in the size and location of the repository footprint also may increase the probability for igneous activity relative to previously published estimates. Using a range of alternative conceptual models, Hill and Stamatakos (2002) described how these uncertainties may have negligible to order-of-magnitude effects on probability model uncertainty.

Number of Waste Packages Affected by Eruption: High Significance to Waste Isolation

The consequences of extrusive igneous activity are directly proportional to the number of waste packages intersected by an erupting volcanic conduit. At present, this number is estimated based on observed conduit size at analog volcanoes. Alternative models of how a volcano may interact with repository drifts and develops a conduit could increase the number of entrained waste packages and thus increase the concentration of radionuclides in erupted ash.

Discussion

Figure 4.3.10-2 shows the sensitivity of extrusive igneous activity dose to the number of waste packages entrained in a volcanic eruption. Normally, in the absence of subsurface drifts, volcanoes form roughly cylindrical conduits along the vertical plane of magma ascent. Based on analogy with deposits at active or deeply eroded volcanoes, staff determined that conduit diameters from 5 to 50 m (16 to 160 ft) represent the most likely range of diameters for a potential future eruption at the proposed repository site (NRC, 1999; Doubik and Hill, 1999). In contrast, the DOE considers potential conduit diameters up to 150 m (500 ft), albeit with very low likelihoods of occurrence (e.g., CRWMS M&O, 2000; BSC, 2003). Actively erupting volcanic conduits have high temperatures and large physical stresses that most likely would completely disrupt any waste package directly intersected by the conduit (NRC, 1999; CRWMS M&O, 2000). Thus, both the NRC and DOE have concluded that any waste package entrained in an erupting volcanic conduit would reasonably fail to provide containment and release its contents into the rapidly flowing magma.

Open drifts located at depths of 300 m (1000 ft) could potentially cause magma ascent and flow processes to behave differently than at undisturbed geologic settings. This is because rising magma is a fluid with an overpressure sufficient to fracture and dilate surrounding wall rock. Intersection with a subsurface drift at essentially atmospheric pressure provides a horizontal pathway out of the original plane of vertical magma ascent, allowing flow localization and nonequilibrium expansion of volatiles (NRC, 1999; Woods, et al., 2002). Using the alternative conceptual model from Woods, et al. (2002), magma could potentially flow down an intersected drift and break out at some point away from the point of original intersection. For randomly located points of intersection and break-out and a single drift containing 155 waste packages, an estimated average of 51 waste packages would be located along the alternative flow path. In contrast, a normal, vertical conduit would intersect an estimated average of 4.5 waste packages using the TPA Version 4.1j code. Figure 4.3.10-2 indicates that there is a directly proportional relationship between the number of waste packages entrained and conditional dose (i.e., dose not weighted by the probability of scenario occurrence). This sensitivity appears reasonable, as the mass of HLW potentially entrained is relatively small compared to the mass of magma. It is assumed that HLW is uniformly distributed in the mass of a modeled eruption; thus, HLW behaves as a trace phase in the magma and does not appreciably affect the transport characteristics of a modeled eruption plume (NRC, 1999; CRWMS M&O, 2000; BSC, 2003).

Uncertainties

In addition to alternative conceptual models for the magma-flow pathway, the number of volcanic conduits created during an igneous event also is uncertain. Using vent location information in Hill and Stamatakos (2002) and assuming medium-to-high confidence magnetic anomalies represent buried volcanoes, it is estimated that there are 17 paired and 13 nonpaired volcanoes in the Yucca Mountain region; most volcano pairs occur in alignments of three to five volcanoes. Volcano pairs have an average spacing of 2.0 ± 1.3 km (1.2 ± 0.8 mi). Assuming that there is a uniform probability of one, two, or three volcanoes intersecting the repository during a potential extrusive event, and that the overall eruption character remains unaffected by the number of volcanic conduits, dose increases by approximately a factor of 2 from this process.

> **Number of Waste Packages Damaged by Intrusion**: Medium Significance to Waste Isolation
>
> *The consequences from intrusive igneous activity are directly proportional to the number of waste packages damaged by direct magma flow into intersected drifts. Damage to waste packages likely occurs from the high thermal, mechanical, and chemical stresses created by basaltic magma. Although process models for these effects have not been developed, available information suggests current waste package design may not provide the physical integrity necessary for waste isolation after direct contact with basaltic magma.*

Discussion

At depth of 300 m (1000 ft), rising basaltic magma is a mixture of gas and melt with sufficient overpressure to fracture and dilate surrounding rock to apertures of approximately 1-m (3-ft) wide. If this confined fluid encounters an open or partially backfilled drift at essentially atmospheric pressure, it will preferentially flow into this drift. Depending on the amount of gas expansion, unobstructed flow speeds can range from on order of 10 m/s (30 ft/s) (Lejeune, et al., 2002) to potentially on order of 100 m/s (300 ft/s) (NRC, 1999; Woods, et al., 2002). Thus, intersected drifts could rapidly fill with magma. Based on current understandings of waste package responses to high temperature, high mechanical load environments, waste packages in direct contact with magma will most likely lose physical integrity and provide no further protection against subsequent hydrologic flow and transport (NRC, 1999, 2002; BSC, 2003). In addition, the high temperatures and complex reducing-to-oxidizing chemical environment created by an igneous intrusion may alter the waste, which could result in more soluble waste forms than intact spent nuclear fuel.

Figure 4.3.10-3 shows that the estimated dose from igneous intrusion increases by about a factor of 3 when the number of waste packages damaged is increased from a base value of approximately 300 to a 95th percentile value of more than 900. Earlier analyses (Figure 4.3.10-4) have shown similar sensitivity of the calculated igneous intrusion dose to the number of waste packages damaged. These sensitivity analyses also do not account for uncertainties in the amount of possible damage to waste packages not directly contacted by magma, but still exposed to potentially corrosive volcanic gasses. Nevertheless, these results indicate that the consequences from releases of radionuclides to the ground water from igneous intrusion are directly proportional to the number of waste packages intersected by magma in a drift. More waste packages damaged by an igneous intrusion leads to higher waste concentrations in the ground water at the RMEI location. However, given the low probability of occurrence for this event, the estimated risk due to effect of magma on a large numbers of waste packages is expected to be low [i.e., probability weighted dose is approximately 0.0001 mSv/yr (0.01 mrem/yr) at 10,000 years for 900 damaged waste packages] (Figure 4.3.10-3).

<u>Uncertainties</u>

The high temperatures and complex chemical environment created by an igneous intrusion likely alters the waste form, which could affect subsequent solubility and transport processes. These may be significant depending on uncertainties in the appropriate model for waste dissolution and near-field transport.

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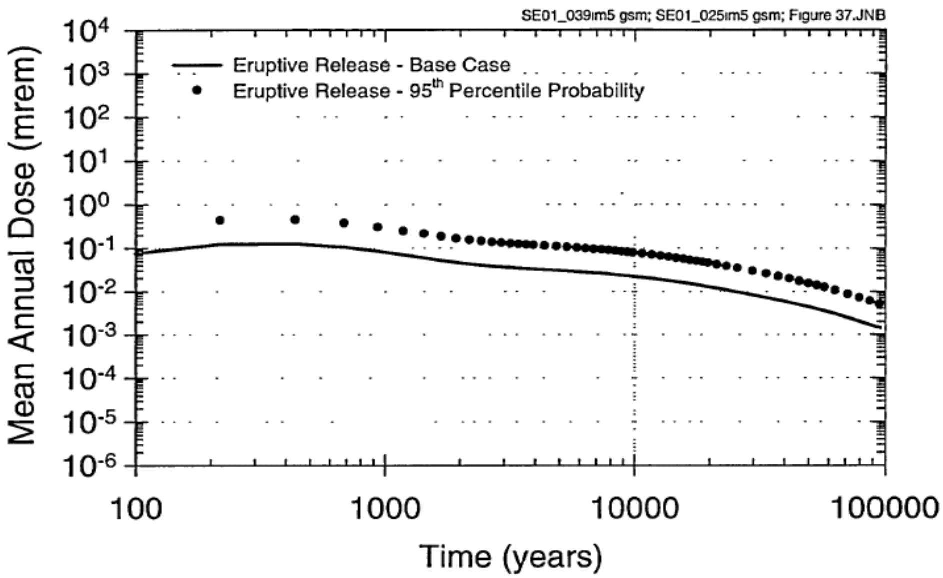


NOTE: Each mean annual dose curve is a probability-weighted average. However, the results of the sensitivity studies do not correspond to expected risk (see introduction to Section 3).

Figure 4.3.10-1. Sensitivity of mean annual dose to igneous activity probability. (From BSC, 2002, Figure 37. Note that dose estimates for variations from the base case do not represent variations in expected risk because the probability of the variation is not accounted for.)

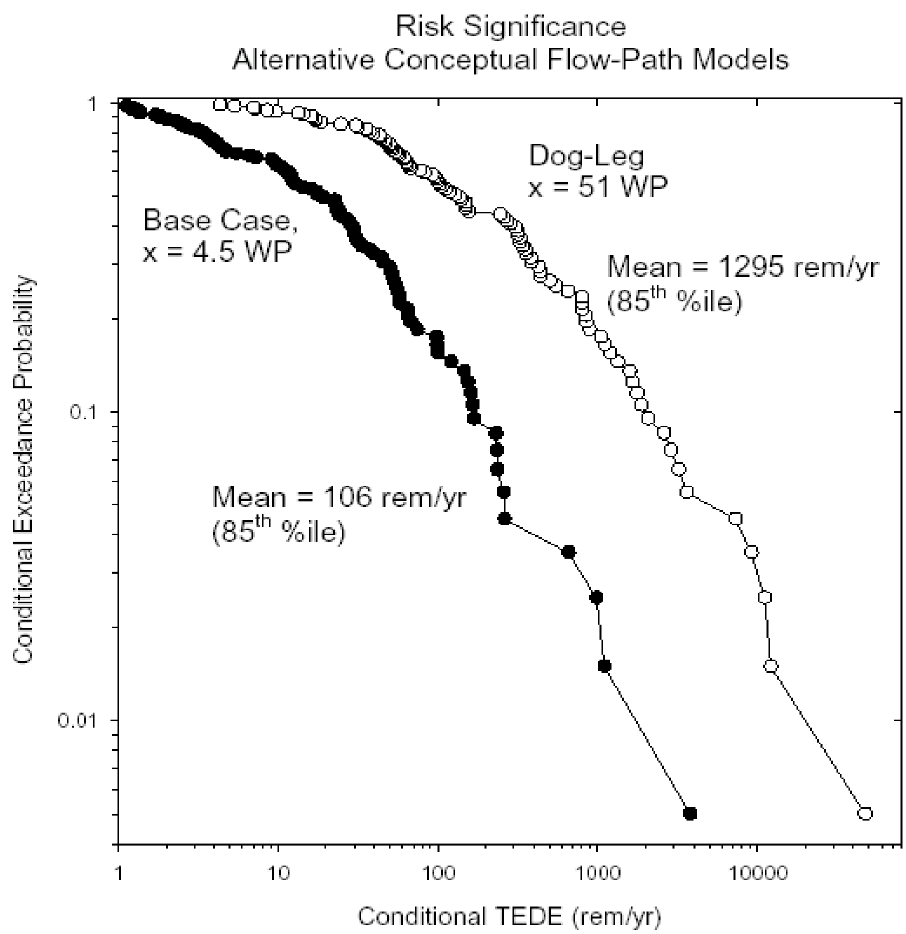
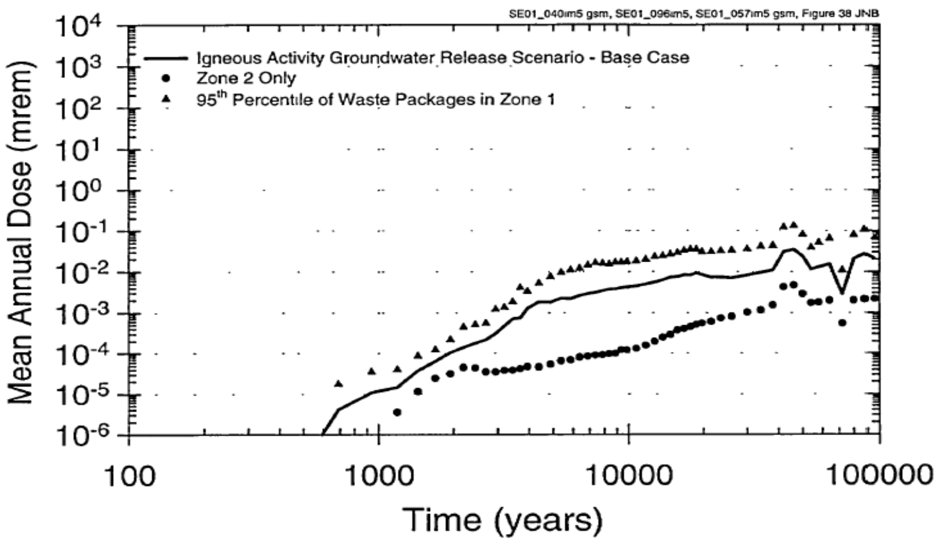


Figure 4.3.10-2. Sensitivity of number of waste packages affected during an extrusive igneous event. [waste packages (WP); total effective dose equivalent (TEDE)]



NOTE: Each mean annual dose curve is a probability-weighted average. However, the results of the sensitivity studies do not correspond to expected risk (see introduction to Section 3).

Figure 4.3.10-3. Sensitivity to number of waste packages hit during an intrusive igneous event. (From BSC, 2002. Note that dose estimates for variations from the base case do not represent variations in expected risk because the probability of the variation is not accounted for.)

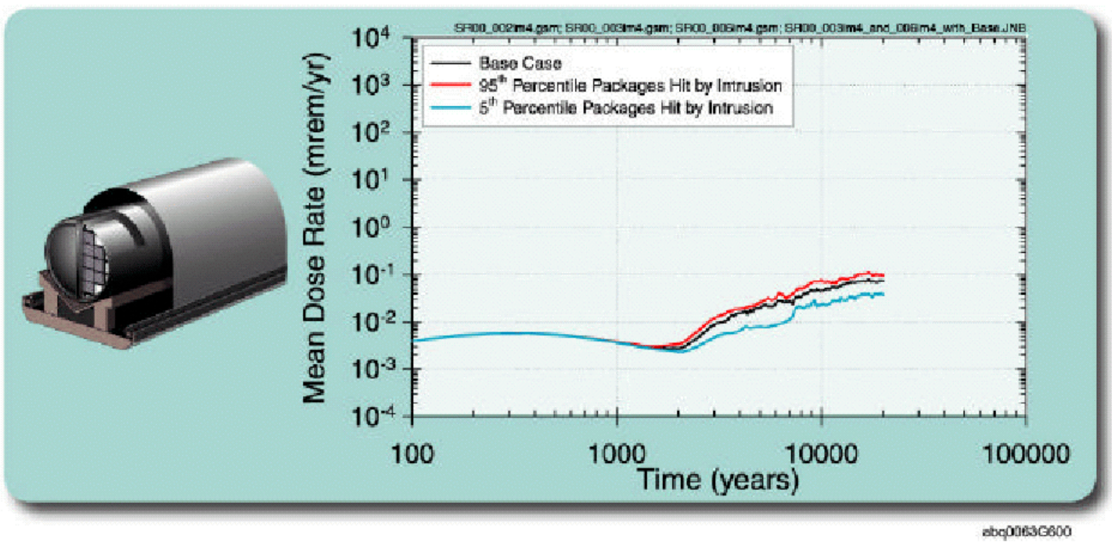


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Figure 4.3.10-4. Sensitivity to the number of waste packages hit during an intrusive igneous event. (From CRWMS M&O, 2000, page F5-46)

4.3.11 Airborne Transport of Radionuclides (DIRECT2)

Risk Insights:	
Volume of Ash Produced by an Eruption	Medium Significance
Remobilization of Ash Deposits	Medium Significance
Inhalation of Resuspended Volcanic Ash	High Significance
Wind Vectors During an Eruption	Medium Significance

4.3.11.1 Discussion of the Risk Insights

Volume of Ash Produced by an Eruption: Medium Significance to Waste Isolation

The concentration of radionuclides in ash is affected by the volume of ash released during an igneous event. Relative to small-volume eruptions, larger-volume eruptions dilute the concentration of HLW in the volcanic deposit.

<u>Discussion</u>

Basaltic volcanoes in the Yucca Mountain region have many characteristics of basaltic cinder cones that have erupted with historical observations. Although most eruption deposits from volcanoes in the Yucca Mountain region are poorly preserved, sufficient information exists to conclude that the range of past activity at these volcanoes is analogous to that observed at historical eruptions (e.g., Connor, 1993; NRC, 1999). Comparison of Yucca Mountain basaltic volcanoes to historical volcanoes with magmatic water contents of at least 2 weight percent shows the ubiquity of an eruption plume that deposits tephra for tens of kilometers away from the vent (e.g., NRC, 1999). Erosion has removed most of the tephra-plume deposits from Yucca Mountain volcanoes; thus, these volumes need to be estimated. The NRC (1999) used deposit ratios from well characterized historical eruptions to estimate volumes of tephra deposits for Yucca Mountain volcanoes, with a similar approach adopted in CRWMS M&O (2000).

The current TPA code uses a relationship between eruption power and duration to calculate ash volume during an eruption. The power and duration ranges used to represent potential igneous events correspond to estimated ash-volume ranges of 6×10^5 to 3×10^8 m^3 (2×10^7 to 1×10^{10} ft^3), with an average volume of 3×10^7 m^3 (1×10^9 ft^3). For comparison, the ash volume for Lathrop Wells volcano is estimated at 5×10^7 m^3 (2×10^9 ft^3) (NRC, 1999). The DOE currently uses a range of ash volumes from 2×10^6 to 4.4×10^8 m^3 (7×10^7 to 2×10^{10} ft^3), with an average volume of 1×10^8 m^3 (4×10^9 ft^3). The effect of these different volume ranges is shown in Figure 4.3.11-1. In this analysis, a factor-of-2 increase in average ash volume resulted in a factor-of-3 decrease in average conditional dose (i.e., dose not weighted by the probability of scenario occurrence).

<u>Uncertainties</u>

Because most of the ash deposits have been eroded from old volcanoes in the Yucca Mountain region, ash volumes for these volcanoes are uncertain. Ash-to-cone volume ratios at historical analog volcanoes can range from approximately 1:1 to 6:1 (NRC, 1999); ratios of 1:1 to 2:1

were used in the NRC (1999) estimates for Yucca Mountain volcanoes. In addition to the presented analyses for areal concentration of entrained waste at 20 km (12 mi), the eruption volume also will affect the potential source-term for remobilization modeling. Although smaller tephra volumes can result in relatively higher initial waste concentrations at 20 km (12 mi), the amount of material available for subsequent remobilization to the 20-km (12-mi) location may be significantly less than for larger volume eruptions. Thus, larger volume eruptions, which may produce deposits with initially lower waste concentrations at 20 km (12 mi), could provide a larger amount of material that would be available for remobilization over time. Remobilization may result in the accumulation of tephra at the 20-km (12-mi) location that is equivalent to or greater than the thickness or concentration of the initial eruption deposit. Both the concentration of radioactive material in air and inhalation dose are sensitive to the deposit thickness and waste concentration in the deposit. As the deposit at the 20-km (12-mi) location evolves through time, remobilization processes could increase the probability-weighted expected annual dose at a time significantly (i.e., tens of years) after the initial eruption. However, current dose estimates, which assume a southerly wind direction, are dominated by the dose occurring in the year immediately following the eruption.

Remobilization of Ash Deposits: Medium Significance to Waste Isolation

After a potential eruption, contaminated ash could be deposited over hundreds to perhaps thousands of square kilometers (tens to perhaps hundreds of square miles). Through time, some of this ash can be erode and transported by wind and water, with later deposition at or near the RMEI location. An influx of remobilized ash could affect the airborne mass loads at the RMEI location, depending on the rate of remobilization and dilution with existing soils.

<u>Discussion</u>

For a potential volcanic event within the repository footprint, most simulated eruptions would deposit some amount of volcanic ash on slopes with drainages that eventually feed into the RMEI location. Through time, wind and water will erode some fraction of the ash deposit and transport it southwards down Fortymile Wash toward the RMEI location. Although tephra-fall deposits can erode within decades from areas with steep topographic gradients, deposits on relatively flat-lying areas are more resistant to erosion (e.g., Segerstrom, 1960). Sediment residence times in the confined channel of Fortymile Wash could be relatively short. Bed-load transport will move sediment down the main channel of the wash during periods of high water flow. In the RMEI area, the Fortymile Wash drainage morphology changes from a steep-sided channel to a broad, braided fan system. This location represents the point where significant long-term sediment deposition occurs within the Fortymile Wash drainage system. Sediment deposition and alluvial aggradation continues south into the Amargosa Desert and overlaps the RMEI location. Consequently, there is likely an initial period of enhanced tephra remobilization before sediment transport rates drop back to pre-eruption values.

The risk significance of remobilization is uncertain. Using a simple mass redistribution relationship, Hill and Connor (2000) suggested that remobilization could increase the net amount of ash at the general RMEI location by a factor of 2 to 10, relative to the original mass deposited by an eruption. This analysis also indicates that, if the wind is directed away from the RMEI during a simulated eruption (i.e., no deposition and, thus, no dose the year immediately following the event), the effect of ash remobilization could result in a dose at some time after the eruptive event at the RMEI location.

Current TPA calculations assume the potential eruption plume is always directed at the RMEI location as a means to account for post-eruption remobilization. These calculations, however, assume that airborne mass loads above ash deposits decay after a potential eruption and that the ash deposit undergoes leaching and erosion with no influx of new material from remobilization. A relatively straightforward approach to evaluating potential risk significance of the remobilization issue is to examine the effect of sustaining airborne particle concentrations at post-eruption values. This effect can be simulated in the TPA Version 4.1j code by slowing the reduction in the airborne mass load with time (i.e., using larger values for the half-life of this process). Larger values represent slower decreases in airborne mass loads from the presumed influx of resuspendable ash through remobilization. Figure 4.3.11-2 shows the relative sensitivity of the decay function parameter in the average conditional dose for 100 realizations of an eruption occurring 1000 years after repository closure. As a proxy for risk significance, the conditional doses for each year from 1000 years to 2000 years are individually weighted by a 10^{-7} annual probability of occurrence and summed. Compared to the risk proxy for a 14-yr half-life, half-lives of 143 yr and 1430 yr result in increases by factors of approximately two and five, respectively.

Uncertainties

Remobilization processes are not well-understood, and supporting data are sparse. Nevertheless, the airborne mass load for the years after a potential volcanic eruption is a highly sensitive parameter in TPA calculations, and uncertainties in this parameter strongly affect calculations of expected annual dose. However, tephra remobilized as a result of surface water is expected to mix with other soils, and transport of tephra by water is expected to result in reduced mass loading, relative to the air transport of tephra during the eruption.

Inhalation of Resuspended Volcanic Ash: High Significance to Waste Isolation

Inhalation of resuspended volcanic ash dominates the total dose for the igneous scenario. Thus, assumptions regarding the amount of fine ash particles in the air significantly influence the calculated dose. The thickness of the deposited ash layer and extent of potential mixing with the underlying soil affects the proportion of ash in the airborne particle load.

Discussion

The amount of fine ash particles resuspended above a deposit depends on the type and duration of surface-disturbing activities and on thickness of the deposit available for entrainment. Based on sensitivity studies using the NRC TPA Version 4.1 code, the parameter for the airborne particle concentration (mass load) above a fresh ash deposit was identified as the most influential to igneous activity dose (Mohanty, et al., 2004). The inhalation dose from a volcanic eruption increases or decreases according to the airborne mass load of waste. The decrease in total mass load after an eruption is assumed to follow an exponential decay in the model. The fraction of contaminated ash in the mass load also can be decreased by mixing of ash with underlying uncontaminated soil. The amount of dilution depends on the thickness of the ash deposit and depth of the surface layer available for resuspension. In undisturbed areas, the resuspension layer is relatively thin [3 mm (0.1 in) in TPA Version 4.1j]; activities such as agriculture disturb a thicker surface layer, and dilute the ash content of the mass load where the thickness of the disturbed layer exceeds that of the ash deposit. The DOE analyses using deeper surface layers [10 mm (0.4 in) and 150 mm (6 in)] lead to lower estimated annual doses that decrease with increasing surface layer thickness. To evaluate the sensitivity of the soil-mixing depth, thickness of the mixing zone was set to 150 mm (6 in), with all other parameters sampled at default values. Figure 4.3.11-3 shows that a factor-of-50 increase in the soil mixing depth results in a factor-of-12 reduction in average conditional dose.

Uncertainties

Further uncertainties exist for appropriate mass loads under different conditions local to the RMEI (e.g., extent and degree of disturbance, indoor or outdoor activities). Use of a soil-mixing zone may not be appropriate for an RMEI that has only a minor component of agricultural habits and only limited surface-disturbing activities. Mass loads from semi-arid regions may not accurately represent appropriate mass loads for the RMEI during the period of peak calculated risk (i.e., first 1000 years postclosure), and many arid terrains may not have soil or vegetation conditions reasonably analogous to the RMEI location. The rate at which mass loading may decrease in the years following an eruption is also uncertain because of complex interrelationships between deposit erosion and the redistribution of inhalable particles. The upper bound of this uncertainty, however, does not appear to affect risk estimates significantly.

Wind Vectors During an Eruption: Medium Significance to Waste Isolation

Both wind speed and wind direction affect the transport of contaminated ash from the eruption source to the location of the RMEI. Wind speed has been shown to be an influential parameter in the sensitivity studies conducted with performance assessment codes. A distribution of wind speeds appropriate to model eruption columns 2 to 7 km (1.2 to 4.3 mi) high needs to be considered. The current TPA approach also fixes the wind direction toward the RMEI to simulate potential effects of post-eruption ash remobilization.

Discussion

In modeling potential volcanic eruptions, the TPA Version 4.1j code uses an exponential distribution of wind speeds with an average of 12 m/second (m/s) [27 mi/hour (mi/hr)], based on limited data. Further analysis of 28,000 measurements from 0 to 7 km (0 to 4.3 mi) altitude at the National Oceanic and Atmospheric Administration Desert Rock Airstrip suggest that a lognormal distribution with roughly the same median value is more appropriate. Calculations using this distribution give doses similar to those computed with TPA4.1j (Figure 4.3.11-4). Greater wind speeds yield proportionally greater dose, presumably because of thicker ash deposits at the RMEI site. In the DOE’s TSPA, setting the wind speed to the 95th percentile value [23 m/s (51 mi/hr)] gives roughly twice the dose as the base case median wind speed of 11 m/s (25 mi/hr) (Figure 4.3.11-5).

Variations in wind direction during an eruption have not been fully analyzed. In both the DOE’s TSPA and TPA4.1j, wind direction was fixed toward the RMEI site to compensate for the lack of any post-eruption movement of contaminated ash. Clearly, if the wind direction is allowed to vary over a realistic range and the potential effects of ash remobilization are ignored, many TPA realizations will not deposit ash at the RMEI location. Scoping analyses presented in Hill and Connor (2000), however, indicated that long-term remobilization processes could result in ash deposits that exceed the thickness of primary volcanic deposits. Calculations that allow wind direction to vary without accounting for potentially significant effects of ash remobilization therefore provide limited insight on risk significance. Because of a lack of information on potential ash remobilization, a medium-risk significance is given to developing an appropriate representation of a realistic wind field above Yucca Mountain.

Uncertainties

The level of detail necessary to reasonably represent a complex wind field is uncertain, given the short transport distances being modeled relative to typical volcanic plume or particle modeling. Variations in deposit thickness on scales of less than a kilometer may be significant to dose calculations, if a realistic wind field and remobilization modeling are used. The time an erupted tephra particle remains at the top of the tephra plume is significantly longer than its rise time from the vent, or its depositional fallout time from the plume. Wind speeds are generally faster at higher altitudes; thus, realistic modeling must consider wind velocity profiles for rapid particle rise, extended lateral advection at the top of the plume, and depositional fallout through gravitational settling. Modeling assumptions (e.g., wind direction fixed in a southerly direction) and sensitivity analyses (e.g., variation of wind speed) have been used to understand the effects of many of these uncertainties.

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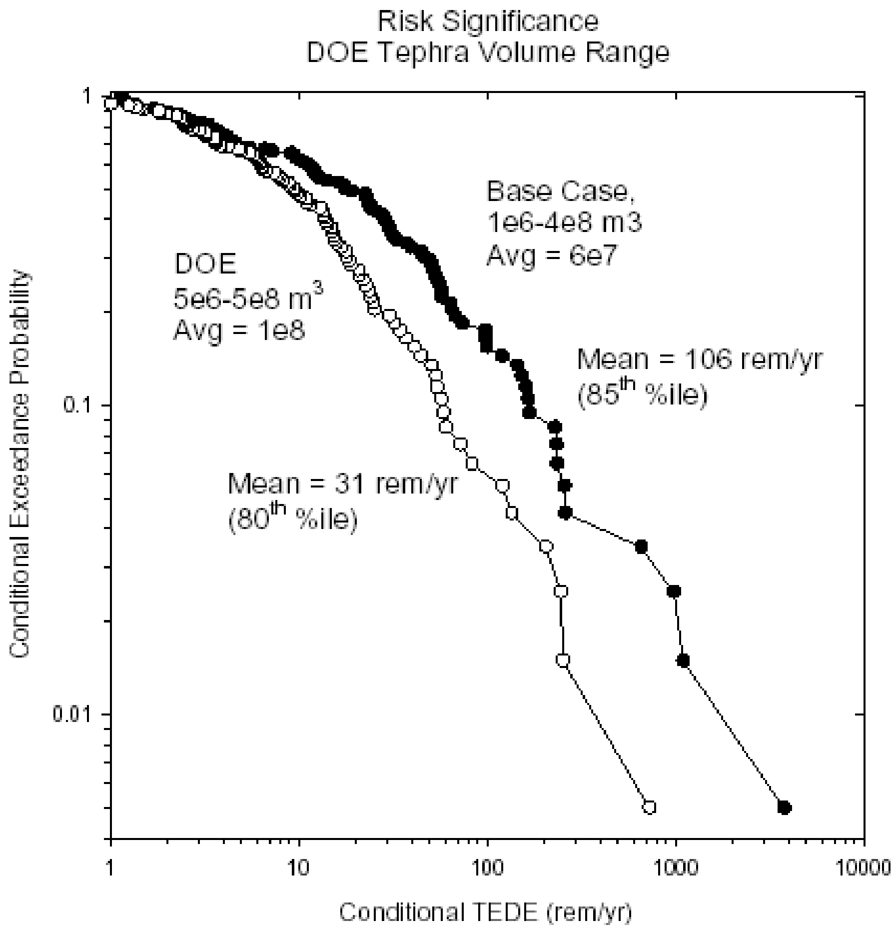

Figure 4.3.11-1. Eruptive volume sensitivity.
[U.S. Department of Energy (DOE); total effective dose equivalent (TEDE)]

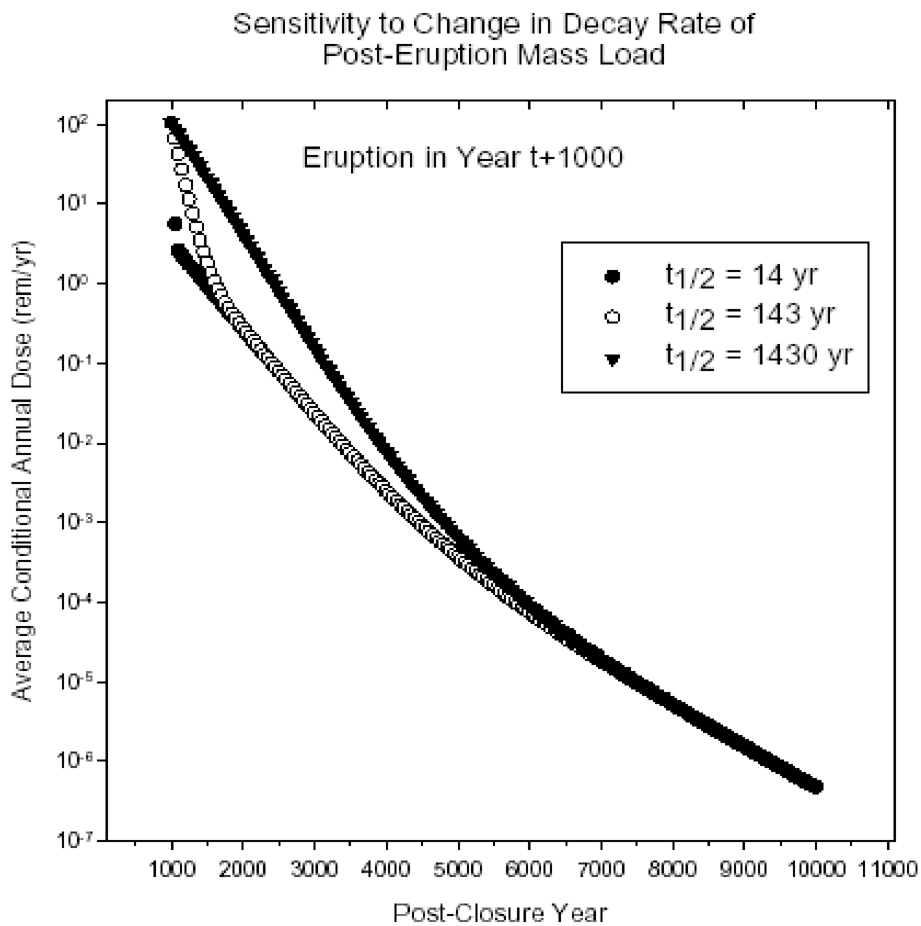

Figure 4.3.11-2. Relative sensitivity for assumptions of airborne mass-load decay function. [time (t); half life ($t_{1/2}$)]

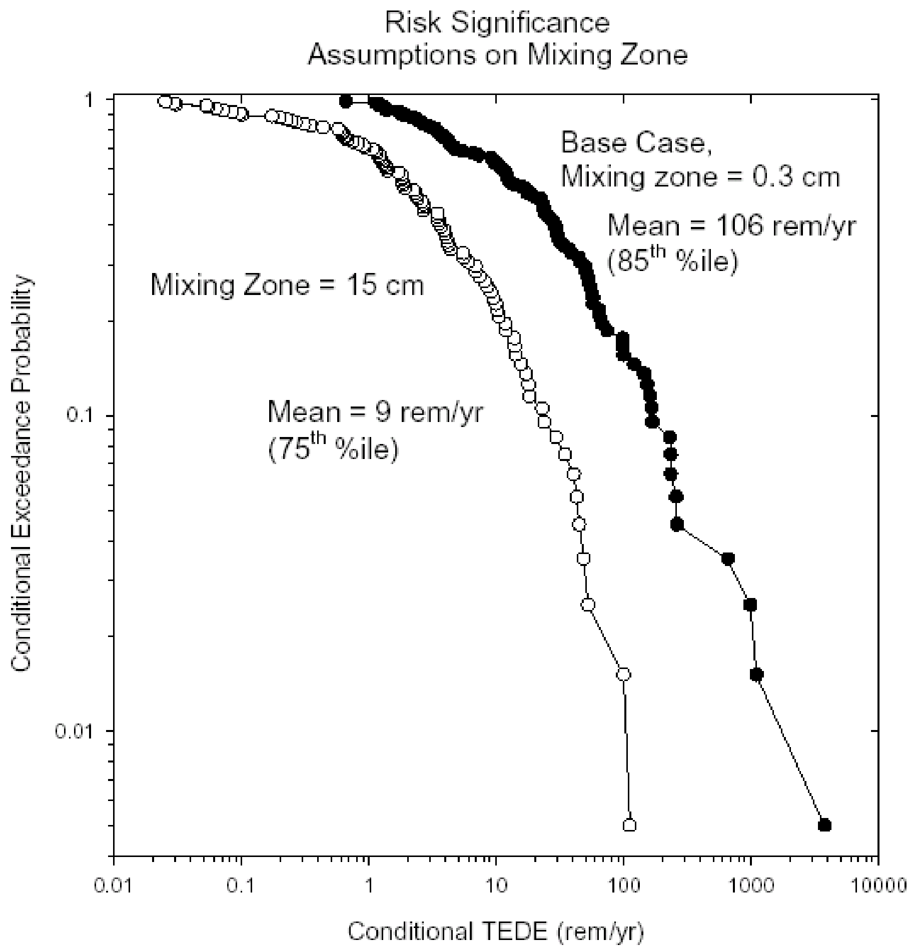

Figure 4.3.11-3. Sensitivity to soil mixing depth.
[total effective dose equivalent (TEDE)]

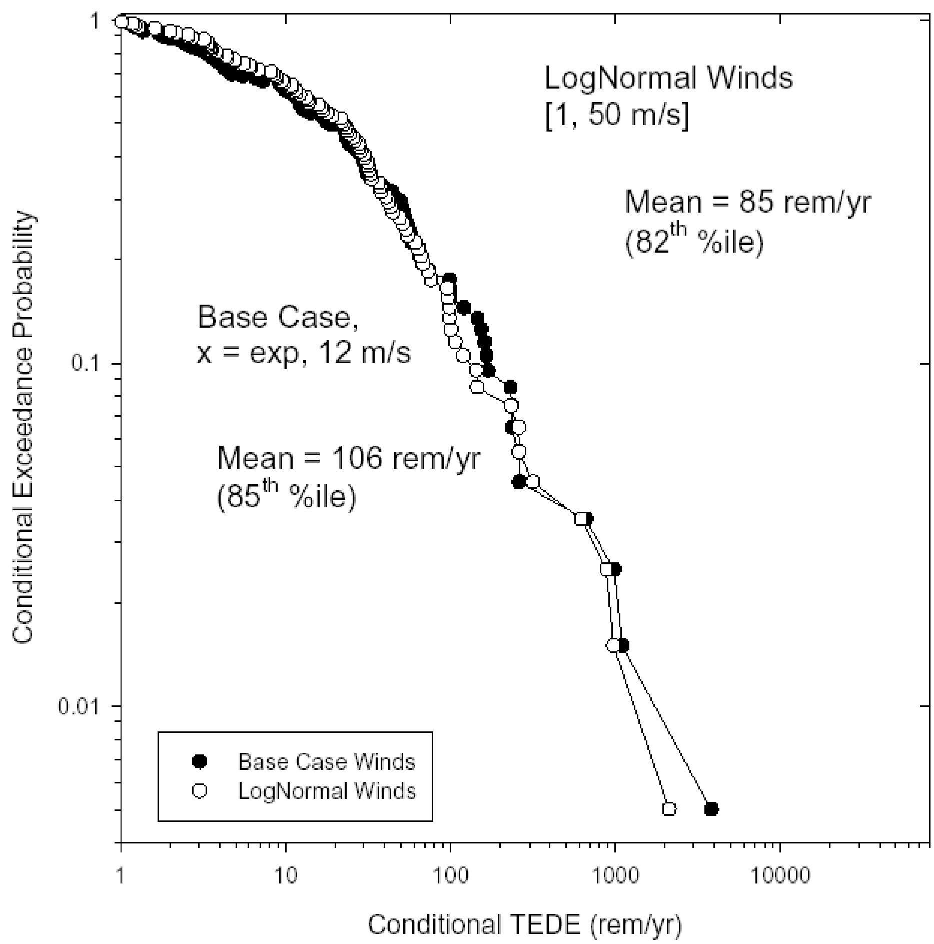

Figure 4.3.11-4. Variations in conditional annual dose using an alternative wind speed distribution. [total effective dose equivalent (TEDE)]

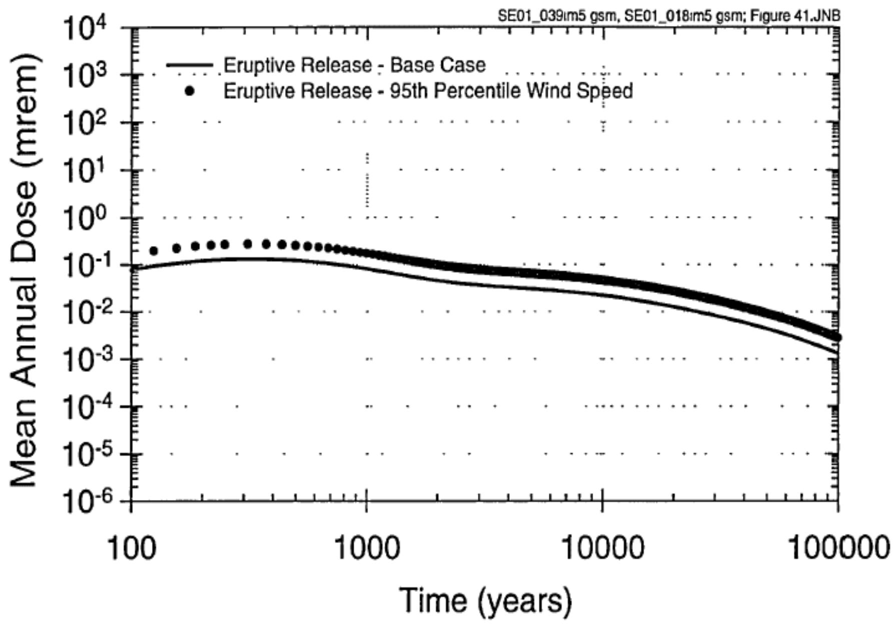

Figure 4.3.11-5. Sensitivity to higher wind speed.
(From BSC, 2002. Note that dose estimates for variations from the base case do not represent variations in expected risk because the probability of the variation is not accounted for.)

4.3.12 Concentration of Radionuclides in Ground Water (DOSE1)

Risk Insights:	
Well-pumping Model	Low Significance

4.3.12.1 Discussion of the Risk Insights

Well-pumping Model: Low Significance to Waste Isolation

In the current well-pumping model, all radionuclides that enter the accessible environment are assumed to be captured in the volume of ground water projected to be withdrawn annually. This assumption limits the risk significance of modeling radionuclide concentrations in ground water.

Discussion

This abstraction relates to estimating the impacts of well pumping on the concentration of radionuclides in water. To limit speculation, this stylized calculation is described in 64 FR 8646 and its implementation is constrained by requirements at 10 CFR Part 63. The calculation involves dividing the estimate of the annual amount of radionuclides, entering the accessible environment, that are captured by the pumping well (or wells), by the volume of water assumed to be pumped to the surface. The annual amount of radionuclides that enter the accessible environment is the result of the release and transport calculations in previously discussed model abstractions, so the risk insights for those abstractions will not be repeated here. The remaining parameters in the concentration calculation do not vary and, therefore, do not have any potential to increase or decrease the resulting concentration. For example, the annual water demand (i.e., pumping volume) is specified by regulation, at 10 CFR Part 63, as 3.7×10^6 m^3 (3000 acre-ft), and all the radionuclides that enter the accessible environment are assumed to be captured in this specified water demand (a conservative assumption).

Uncertainties

No variation or uncertainty is generated in this abstraction because the regulation at 10 CFR Part 63 sets the pumping volume as 3.7×10^6 m^3 (3000 acre-ft) and all radionuclides in the plume are conservatively assumed to be captured by the pumping well.

4.3.12.2 References

None.

4.3.13 Redistribution of Radionuclides in Soil (DOSE2)

Risk Insights:	
Redistribution of Radionuclides in Soil	Low Significance

4.3.13.1 Discussion of the Risk Insights

Redistribution of Radionuclides in Soil: Low Significance to Waste Isolation

Ground-water-based dose estimates are primarily influenced by the drinking water pathway, thereby limiting the importance of pathways related to radionuclides in soil. Igneous-activity-based dose estimates are dominated by inhalation of radionuclides that have low mobility in soil, so leaching processes do not significantly affect estimated doses (low soil leaching leads to higher crop ingestion doses).

Discussion

The model abstraction for redistribution of radionuclides addresses the movement of radionuclides after deposition on the ground, either through surface application of ground water or settling of volcanic ash after an eruption. Redistribution affects the quantity and concentrations of radionuclides accessible to human receptors in the biosphere, and therefore, influences the dose estimates from radionuclides deposited on the ground. Redistribution can increase exposure if the transport processes involved move material closer to human intake pathways (e.g., resuspension to the breathing zone of an individual) or decrease exposure if transport is away from human exposure pathways (e.g., leaching to deep soil layers) or transport substantially dilutes initial radionuclide concentrations.

For ground-water-based dose estimates, biosphere modeling results (Figure 4.3.13-1) show that for the radionuclides that dominate the current dose estimates (Table 4.3.13-1), the drinking water pathway, which is not affected by soil redistribution processes, would contribute approximately 50 percent of the all-pathway dose estimates. Because only the remaining half of the all-pathway dose can be influenced by redistribution processes and this portion of the dose is dominated by the crop-ingestion pathway (Figure 4.3.13-1), the effect of redistribution processes on the all-pathway dose is limited. In the biosphere model, crops can become contaminated through root uptake or deposition of resuspended material. As a result, redistribution processes that alter the soil concentration on the soil surface and in the root zone of the crops can affect the crop-ingestion dose. These processes include leaching of contaminants to deeper soil layers away from roots, and buildup of contaminants from irrigation. Any potential impacts from contaminants leaching from the soil to the ground water are not addressed by the current model. Such secondary-use consequences are assumed to be lower than consequences attributed to initial RMEI use, because of the attenuating effects of dilution during transport.

To test the impacts of soil leaching on dose-modeling results, the most variable parameter in the leaching calculation--the distribution coefficient--was input at the extremes of the range used in TPA Version 4.1d biosphere calculations. The results (Figure 4.3.13-2) indicate that the greatest potential change in dose from variation in this parameter is about a factor of 5. Because it is unlikely that the value of every distribution coefficient would be at the highest value of its known range, the effect on dose estimates from more realistic changes to this parameter is expected to be far less than the factor of 5 and is therefore considered of low-risk significance. This conclusion is further supported by the results of a system-level sensitivity analysis (Mohanty et. al., 2004) that found no consistent significant influence on dose from soil-leaching parameters when all other total-system model parameters were sampled.

The DOE analyzed effects of soil buildup on biosphere dose-modeling results (CRWMS M&O, 2000) by modeling irrigation for time periods sufficient for soil concentrations to reach equilibrium (e.g., soil concentration remains constant with time). Results suggest the dose results for most radionuclides would be expected to change by 15 percent. Some radionuclides (i.e., americium, cesium, nickel, protactinium, plutonium, radium, strontium, thorium, uranium) showed changes above this level (CRWMS M&O, 2000); however, these radionuclides are not contributing to the ground-water-based dose estimates. In general, the properties that lead to buildup in soil (e.g., low mobility) also favor slow transport times in ground water.

For the igneous activity dose calculations, both the NRC (Figure 4.3.13-3) and DOE (BSC, 2001) results indicate that the dose is dominated by inhalation of resuspended contaminated ash deposited from an eruption. Both the NRC (Figure 4.3.13-4) and DOE (CRWMS M&O, 2000b, Figure 4.2-3) analyses indicate over 90 percent of the direct-release dose are from radioactive species of the elements americium and plutonium. The chemical properties of these elements lead to low leaching in soils, as indicated by the data and related information presented in Sheppard and Thibault (1990). A simple quantification of the low-leaching effect, using the environmental deposition and removal calculation described in the GENII v1.485 user manual (Napier, et al., 1988) and leaching factors calculated in the TPA code for plutonium and americium, indicates that the annual surface soil concentration is reduced by less than 1 percent when leaching to deeper soil layers is considered.

<u>Uncertainties</u>

For the ground-water-release biosphere calculations, leaching of radionuclides in soils is an uncertain process. However, the aforementioned analyses suggest the magnitude of the impacts of this uncertainty on dose is low when evaluated in the context of other uncertainties in the performance assessment (i.e., the variation in the biosphere calculations is small compared to the rest of the performance assessment). For the igneous release, the uncertainty in the leaching behavior is less important because radionuclides that dominate the dose have low mobility in soils. Other potential redistribution processes (e.g., surface remobilization) are somewhat uncertain.

4.3.13.2 References

BSC. "FY01 Supplemental Science and Performance Analyses, Volume 1: Scientific Bases and Analyses." TDR-MGR-MD-000007, Revision 00, ICN 01. pp. 13-44. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001.

CRWMS M&O. "Abstraction of BDCF Distributions for Irrigation Periods." ANL-NBS-MD-000007 Rev 00 ICN 01. Las Vegas, Nevada: Civilian Radioactive Waste Management System Management and Operating Contractor. 2000.

CRWMS M&O. “Total System Performance Assessment for the Site Recommendation.” TDR-WIS-PA-000001 REV 00 ICN 01. Las Vegas, Nevada: Civilian Radioactive Waste Management System Management and Operating Contractor. 2000b.

Napier, B.A., R.A. Peloquin, D.L. Strenge, and J.V. Ramsdell. “GENII: The Hanford Environmental Radiation Dosimetry Software System, Volume 1: Conceptual Representation.” PNL-6584. Volume 3. Richland, Washington: Pacific Northwest Laboratory. 1988.

Sheppard, M.I. and D.H. Thibault. “Default Soil Solid/Liquid Partition Coefficients, Kds, for Four Major Soil Types: A Compendium.” Health Physics. Volume 59. Number 4. pp. 471-482. New York, New York: Pergammon Press. 1990.

Table 4.3.13-1. Primary radionuclides contributing to peak expected dose. (From Mohanty, et al., 2004, Table 3-13)

	10,000 Years		100,000 Years	
Radionuclide	**Mean Value Data Set (mSv/yr)**	**Multiple-Realization Data Set (mSv/yr)**	**Mean Value Data Set (mSv/yr)**	**Multiple-Realization Data Set (mSv/yr)**
Np-237	0	4.29×10^{-5}	3.69×10^{-2}	9.54×10^{-2}
I-129	1.30×10^{-4}	5.34×10^{-5}	3.90×10^{-4}	1.33×10^{-3}
Tc-99	2.15×10^{-4}	1.09×10^{-4}	6.17×10^{-4}	2.09×10^{-3}
U-234	0	1.77×10^{-9}	4.62×10^{-7}	6.80×10^{-5}
Cl-36	7.11×10^{-7}	2.64×10^{-7}	1.35×10^{-6}	5.10×10^{-6}
Se-79	0	3.74×10^{-8}	9.31×10^{-6}	1.14×10^{-5}

Note: [neptunium-237 (Np-237); iodine-129 (I-129); technetium-99 (Tc-99); uranium-234 (U-234); chlorine-36 (Cl-36); selenium-79 (Se-79)]

TPA Base Case Mean Value Run

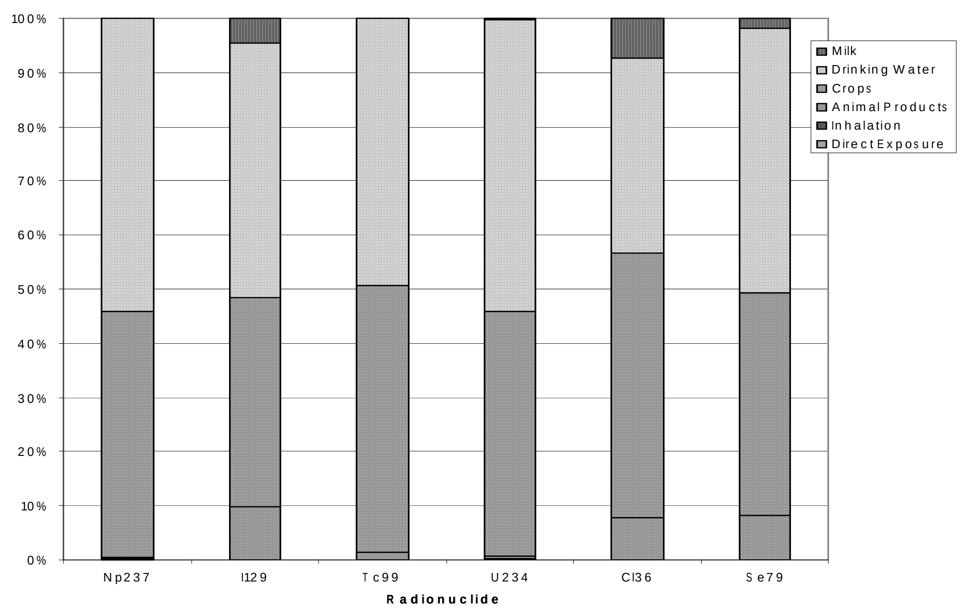

Figure 4.3.13-1. Ground water release scenario: Exposure pathway contributions to dose for important radionuclides (using the TPA computer code). [neptunium-237 (Np-237); iodine-129 (I-129); technetium-99 (Tc-99); uranium-234 (U-234); chlorine-36 (Cl-36); selenium-79 (Se-79)]

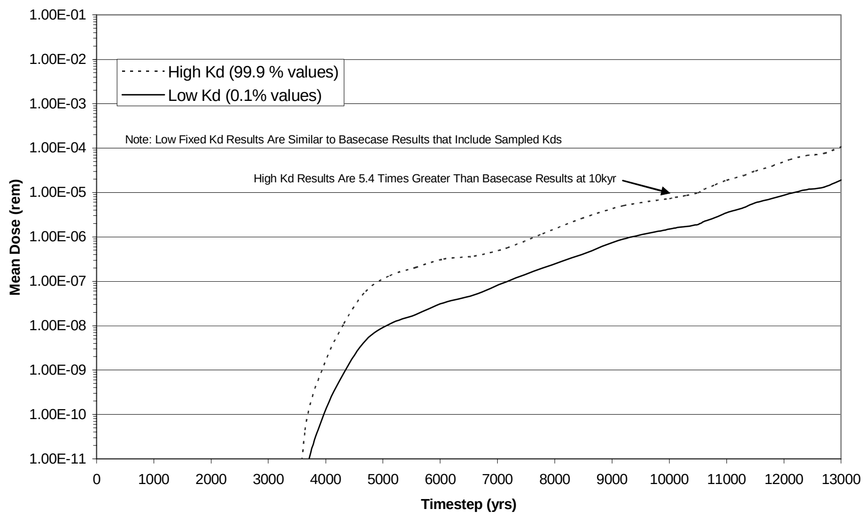

Figure 4.3.13-2. Comparison of dose curves from base case and high/low perturbations of soil distribution coefficients (Kd) using the TPA computer code, Version 4.1d.

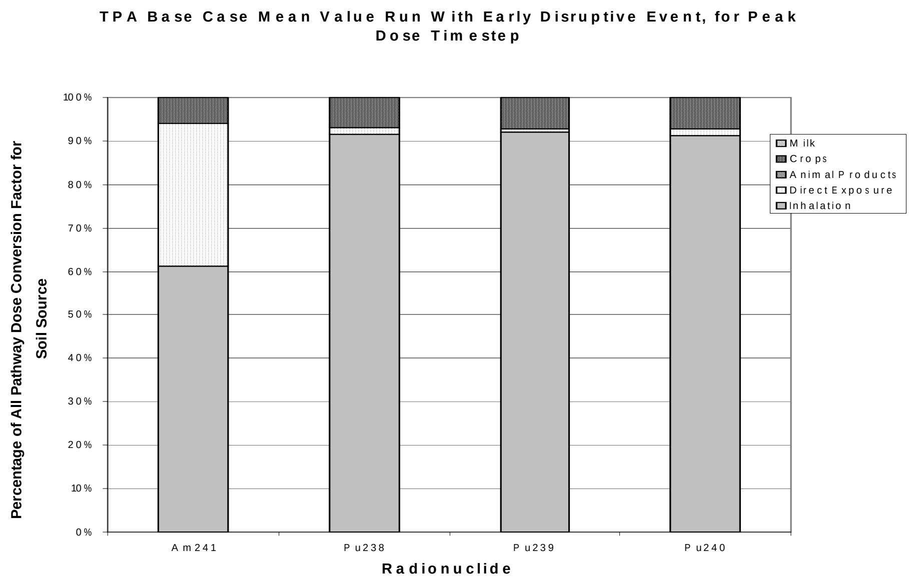

Figure 4.3.13-3. Igneous activity release scenario: Exposure pathway contributions for important radionuclides (using the TPA computer code). [americium-241 (Am-241); plutonium-238 (Pu-238); plutonium-239 (Pu-239); plutonium-240 (Pu-240)]

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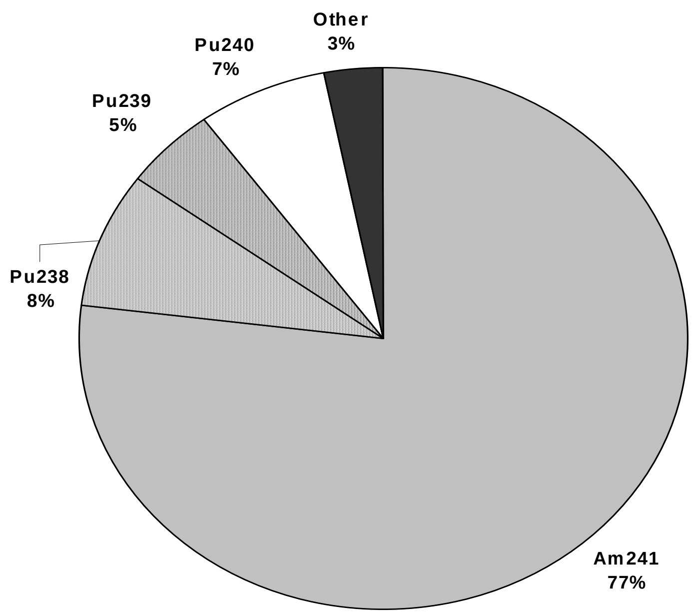

Figure 4.3.13-4. Key radionuclides for igneous activity disruptive event dose (using the TPA computer code). [americium-241 (Am-241); plutonium-238 (Pu-238); plutonium-239 (Pu-239); plutonium-240 (Pu-240)]

4.3.14 Biosphere Characteristics (DOSE3)

Risk Insights:	
Characterization of the Biosphere	Low Significance

4.3.14.1 Discussion of the Risk Insights

Characterization of the Biosphere: Low Significance to Waste Isolation

The regulation at 10 CFR Part 63 specifies mean values to be used for many important biosphere parameters, thereby limiting the effect of biosphere modeling assumptions and parameters on total-system risk estimates.

<u>Discussion</u>

The NRC regulations at 10 CFR Part 63 specify the use of mean values for behavioral input parameters (i.e., diet and living style) such as consumption rates and exposure times. This reduces the range of variation in the ground-water-release biosphere model abstraction calculations (NRC, 2002; page 3.3.14-11). The DOE evaluation of the impact of the biosphere modeling variation on estimated dose results is shown in Figure 4.3.14-1.

<u>Uncertainties</u>

As noted in the discussion, the uncertainties in the biosphere calculations are limited by requirements at 10 CFR Part 63. Based on the available parameter information used for the ground-water-release biosphere dose calculations, the staff does not expect that a significant increase in the uncertainty propagated in the biosphere calculations would occur from additional information. For igneous activity biosphere dose calculations, the modeling of features and processes that lead to resuspension of contaminated volcanic ash (e.g., the mass-loading factor) at the location of the RMEI is both highly uncertain and important to dose results. Although conceptually this is a biosphere abstraction issue, it is also addressed in the igneous disruptive event abstraction in Section 4.3.11 and is not considered further in ranking the significance to waste isolation of the biosphere.

4.3.14.2 References

CRWMS M&O. “Total System Performance Assessment for the Site Recommendation.” TDR-WIS-PA-000001 REV 00 ICN 01. Las Vegas, Nevada: Civilian Radioactive Waste Management System Management and Operating Contractor. 2000.

NRC. NUREG-1762, “Integrated Issue Resolution Status Report.” Washington, DC: U.S. Nuclear Regulatory Commission. July 2002.

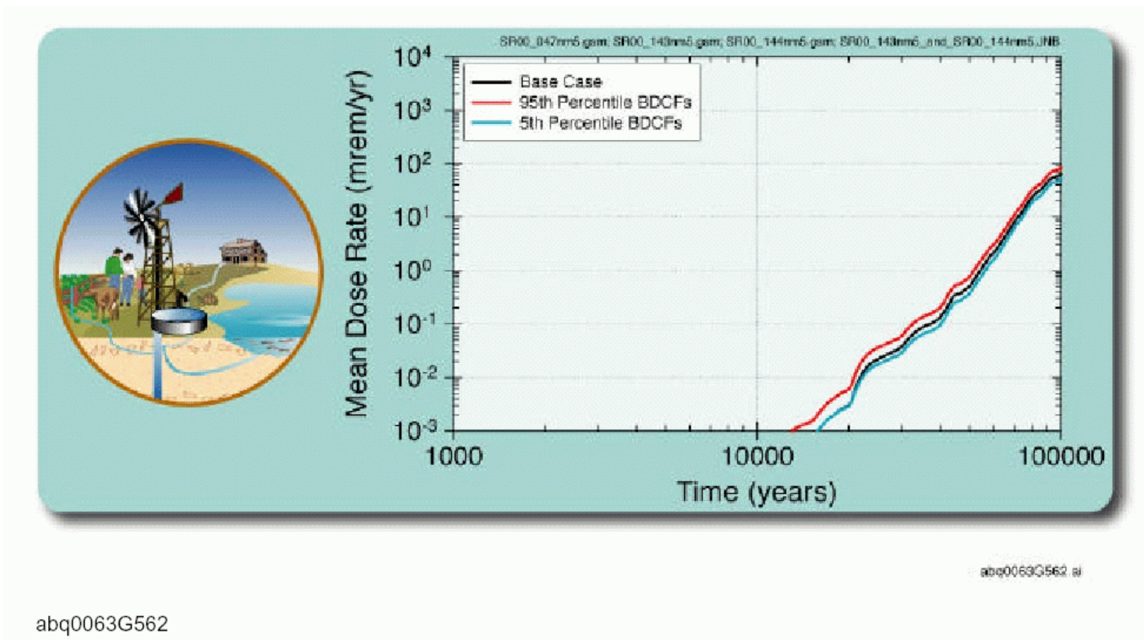

Figure 4.3.14-1. Sensitivity to biosphere dose conversion factors. (From CRWMS M&O, 2000, page F5-38)