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Civilian Radioactive Waste Management System
Management & Operating Contractor

Total System Performance Assessment for the Site Recommendation

TDR-WIS-PA-000001 REV 00 ICN 01

December 2000

Prepared for:

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DISCLAIMER

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CHANGE HISTORY

Revision Number	Interim Change No.	Description of Change
00	00	Initial issue
00	01	Changes throughout as indicated by change bars. Causes for change include DOE comments, typographical errors, and completion of supporting documents.

Revised reference callout format throughout the document and updated the reference list by adding and deleting Document Input Reference System (DIRS) numbers. References for added DIRSs numbers:

100061	148384	153002	153178
100746	148713	153038	153184
101173	148992	153039	153200
103748	149092	153105	153201
105155	149862	153111	153202
122137	149939	153122	153269
131861	151294	153123	
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100065	130997	146099	151252
100066	131951	146104	151293
100362	133420	146376	151347
103445	135968	147120	151547
103805	139610	148449	151715
107538	140418	150532	151718
113534	141440	150824	152207
119414	143368	150826	152209
124151	144335	150924	152217
124314	144454	151160	
130590	144565	151064	

Incorporated updated climate for 1,000,000 year simulations. Incorporated additional analyses of secondary phases effect on performance.

This ICN utilized the FY2000 Technical Development Plan, since it is just a minor update of Rev 00. Technical Work Planning documentation will be developed for Rev 01.

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EXECUTIVE SUMMARY

This document presents the results of the total system performance assessment conducted for the *Site Recommendation Consideration Report* that is currently being developed. This assessment is one of an iterative series of analyses conducted about Yucca Mountain over the life of the Yucca Mountain Site Characterization Project to support the decision of the Secretary of Energy on whether to recommend the site to the President for construction of a geologic repository.

The performance assessment for the site recommendation is used to evaluate the ability of the engineered and natural systems of the geologic repository to isolate nuclear waste for ten thousand years. A separate document, the Environmental Impact Statement considers repository performance for an additional several hundred thousand years. This document will present some of those long-term analyses. The performance assessment analyzes the behavior of the reference design of the engineered repository components in the expected natural conditions at the Yucca Mountain site (nominal scenario). It also evaluates the contribution of the geologic setting to waste isolation and includes sensitivity and uncertainty analyses to illustrate the relative importance of the various components and parameters. Unexpected disruptive events and their effect on performance of the potential repository are also analyzed in the performance assessment (disruptive scenario class).

This document summarizes the performance assessment work performed for the site recommendation considerations report. Readers who would like more information on how the performance assessment was developed and performed should consult the *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]), as well as supporting Process Model Reports and Analysis Model Reports.

Document Organization

Section 1: General Description of the Total System Performance Assessment Process–Section 1 explains in detail what total system performance assessment is and why it is applicable to potential repository development. It also discusses, from the perspective of the international radioactive waste management community, the general approach for performing a total system performance assessment.

Section 2: Yucca Mountain Total System Performance Assessment for the Site Recommendation–This section describes the specific way in which the general performance assessment approach was adapted for the Total System Performance Assessment-Site Recommendation. It describes how the potential repository system is represented in the performance assessment, based on current knowledge of the site. This description traces the eventual release of radionuclides (or nuclide) to the biosphere using the attributes of repository performance. The section then explains the method used to build the computer model, and how uncertainty and variability were treated in the analyses. Finally, it describes the traceability of the information used in the model.

Section 3: Development of Model Components–The total system performance assessment model represents the entire system of repository behavior; this overall model is made up of a series of models or components that represent the processes that are expected to influence system performance. Section 3 presents a detailed description of how each component was developed, and provides some results of analyses for specific aspects of individual components.

Section 4: Performance Analyses–This section describes which aspects of the components were combined into a total system performance model and explains how and why it was done. It then reports modeling results in terms of the nominal case scenario, the disruptive case scenario, and a combined nominal and disruptive case. The modeling results from the stylized human intrusion scenario are also presented, followed by a discussion of potentially disruptive events not included in the analysis. Finally, the section includes an brief comparison of design alternatives to the reference design. The two alternatives analyzed are a case that includes the use of backfill, and a case that utilizes a low-thermal load approach.

Section 5: Sensitivity Analyses for Total System Performance Assessment Components–This section evaluates the importance of uncertainty to the nominal, disruptive, and combined scenarios. It discusses factors in each model about which significant uncertainties exist in the current scientific understanding, and it examines their relative importance to repository system performance. It also examines how sensitive those factors are to changes in the values assigned to them. Lastly, it addresses the robustness of the components of system performance.

Section 6: Summary and Conclusions–In contrast with the previous section, which deals with the uncertainty and sensitivity of various aspects of individual components, this section looks at uncertainty from a total system perspective.

EXPLANATION OF A TOTAL SYSTEM PERFORMANCE ASSESSMENT

The general total system performance assessment process has developed over time through its application on numerous projects by various international organizations involved in radioactive waste management and in consultation with the U.S. Nuclear Regulatory Commission. The TSPA must be based on a thorough understanding of the relevant processess that may affect performance and site-specific information, natural analogs that assist in building the confidence in the long term processes evaluated in the TSPA, and relevant laboratory data concerning the engineered materials. The TSPA approach allows an analysis of the system that appropriately incorporates and quantifies the uncertainty in such a long term projection of repository performance. The TSPA-SR aims to provide a defensible analysis of system behavior incorporating models and parameters that are based on scientific observations in order that decision-makers can assess the ability of the repository system to comply with proposed regulations.

The TSPA process can be visualized as a series of levels going up a pyramid (Figure ES -1). The base of the pyramid is built using all of the data and information collected by scientists and engineers involved in site characterization and engineering design. This information is used to develop appropriate models which describe the features, events, and processes that may be present in the potential repository system. The base is large because it represents the composite of all the information gathered by the repository program.

This information provides the basis for the development and testing of conceptual models. A conceptual model is a set of qualitative descriptions used to describe a system or subsystem for a given purpose. An example is a description of the movement of water molecules as they pass between rock and fractures. There may be several alternative conceptual models that provide a reasonable description of a particular system or subsystem.

The specific aspects for describing a process on a larger scale are then extracted and incorporated into computer models to deal with each of the relevant features, events, and processes. An example is a model for all water flow above the water table, which would incorporate flow interactions between the rock matrix and the rock fractures as well as many other specifics needed to describe how water flows throughout the rock mass. This abstraction or progressive simplification to a more compact and usable form is depicted by the slightly smaller width of the pyramid. The models that eventually analyze the evolution through time of all the various components of the system are generally the most compact or abstracted models of all. These abstracted models start with the results of the detailed process level modeling and create a representation that captures all the salient features of the process model, and the associated uncertainties. Abstraction is necessary for many reasons. One of these reasons is that many of the models are much too large to be run efficiently even on very large computers.

To capture the full detail of the uncertainty and variability in the behavior of the repository system, the total system performance assessment must be probabilistic, using multiple calculations (as opposed to deterministic or a single calculation using a single value for each parameter in the system). The models are run many times using many combinations of parameters. Each of the combinations of parameters has some definite possibility of representing the actual performance of the potential repository. These probabilistic analyses are intended to reflect the range of behaviors or values for parameters that could be appropriate, knowing that perfect or complete knowledge of the system will never be available and that the system is inherently variable.

A final reason to use abstraction is that, in some cases, an overly complex model would over represent the actual state of knowledge about a process, so a simpler model is more appropriate, i.e., the complex model could be more advanced than the data available for a system.

A more detailed depiction of the total system performance assessment process is shown in Figure ES-2. Here, collection of site data and incorporation of the data (or estimates, where data are not available) is illustrated first into conceptual models, then into mathematical equations, next into computer (numerical) models and, finally, into a total system model. The figure is a more detailed representation of the process that is depicted using the total system performance assessment pyramid in Figure ES-1.

How the Potential Repository System Is Visualized in the Total System Performance Assessment-Site Recommendation

In general, the potential repository system is visualized as a series of processes linked together, one after the other, spatially from top to bottom in the mountain. From a computer modeling point of view, it is important to break the system into "bite-size" portions that relate to the way information is collected. In reality, the potential geologic repository system will be completely interconnected, and essentially no one process will be independent of other processes. However, the complexity of the system demands that some idealization of the system be developed for an analysis to be performed.

The overall system, in progressively greater detail from mountain scale down to waste form scale, is shown in Figure ES-3. This figure illustrates several of the key natural and engineered barriers that contribute to the long-term isolation of waste from the biosphere. The natural barriers include semi-arid natural environment and the location of the repository about 300 m beneath the ground surface, and about 300 m above the water table. These natural barriers are enhanced by engineered barriers including the waste package and drip shield.

The attributes of the potential repository performance associated with this conceptualization are shown in Figure ES-4. Each of the attributes and the associated component models are shown on Figures ES-5 to ES-9 in their relative spatial sequence. Each model in the sequence is shown in Figure ES-10 and provides input to the following model and receives the output of the preceding model or models. The shape of the component model icons shown on these figures is determined by the attribute of the potential repository performance.

The attributes of the potential repository performance are the following:

- Limiting water contacting the waste packages
- Prolonging waste package lifetime
- Limiting radionuclide mobilization and release from the EBS
- Slower radionuclide transport away from the EBS
- Low mean annual dose even considering potentially disruptive events and processes.

The disruptive events icons are used to depict the models associated with off-normal or disruptive events such as volcanism. These events, if and when they occur, would affect the nominal case processes. Human intrusion to the potential repository is an additional scenario also evaluated in the TSPA.

The following is an abbreviated description of the expected behavior of the major components.

Limiting Water Contacting the Waste Packages–The changes in climate over time provide a range of conditions that determine how much water falls onto the ground surface and infiltrates into the ground below (Figure ES-5). Based on current scientific understanding including paleoclimate studies, the assumption in the total system performance assessment is that the current climate represents one of the driest climates that the Yucca Mountain site will ever encounter. All future climates are assumed to be either similar to current conditions or wetter. The water that is not lost back to the atmosphere by evaporation or transpiration enters the unsaturated zone flow system. Water infiltration is affected by a number of factors related to the climate state, such as increase or decrease in vegetation on the ground surface, total precipitation, air temperature, and runoff.

Water generally moves downward in the rock matrix and fractures. The rock mass at Yucca Mountain is composed of volcanic rock that is fractured to varying degrees as a result of contraction during cooling of the original nearly molten rock and also due to extensive faulting in the area. Water flowing in the fractures moves more rapidly than the water moving through the matrix. In some locations, some of the water collects into locally saturated zones in the rock or is diverted laterally by differences in the rock properties. The overall unsaturated flow system is heterogeneous, and the location of flow paths and velocities and volumes of groundwater flowing along these paths are expected to change many times over the life of the potential repository system.

The heat generated by the spent nuclear fuel in the potential repository will cause the temperature of the surrounding rock to rise to a peak (or maximum) level within decades to centuries after emplacement and then decay gradually back to ambient temperature over thousands of years. Much of the water and gas in the heated rock will be driven away from the potential repository during this heating period. The thermal output of the waste decreases with time, and as the rock temperature cools, percolating water (including some of the mobilized water) will likely flow back toward the potential repository. Some of the water that contacts the potential repository walls can drip or seep into the potential repository, but only in a relatively few places. The number of seeps that can occur and the amount of water that is available to drip into the drifts is restricted by the low volume of water flowing through Yucca Mountain and by draining through the rock pillars between the drifts. Drips also can occur only if the hydrologic properties of the rock mass cause the water to concentrate enough to feed a seep. Over time, the number of seeps increases and decreases, and their locations change, corresponding to increased or decreased infiltration based on changing climate conditions and on mineralogic changes to fractures. The drips will be directed away from the waste packages for a considerable time due to the protection afforded by the drip shield in the engineered barrier system.

Prolonging Waste Package Lifetime–Because the potential repository is located above the water table in the unsaturated zone, the most important process controlling nominal case waste package lifetime is moisture on the waste package (Figure ES-6), either from seeps or moisture in the air. The location of the seeps providing dripping water depends to some extent on the natural conditions of the rock, but also on the alterations caused by potential repository construction. Alterations, such as increased fracturing, may be caused by mechanical processes related to drilling the drifts or by thermal heating and expansion of the drift wall. The alterations in the seepage can also be caused by chemical alterations enlarging or constricting some of the pores and /or the fractures. This can occur due to evaporating water precipitating minerals, condensing water dissolving minerals, and engineered materials dissolving in water and reprecipitating in the surrounding rock. The chemistry in the drift changes because of the complex interactions among the incoming water, circulating gas, and materials in the drift (e.g., metals in the drift support system, drip shield, or waste package). The chemical evolution is strongly influenced by heat during the period of thermal heating.

In the reference design, the radioactive waste emplaced in the potential repository will be enclosed in a two-layer waste package. The layers will be constructed of two different materials that are expected to degrade at different rates and from different mechanisms as they are exposed to various potential repository conditions. The outer layer will be made of high-nickel alloy metal and the inner layer of a stainless steel. The design also has a drip shield made of titanium. These will be emplaced over the waste packages to reduce the potential for dripping water hitting the waste packages. Where the waste packages are exposed to dripping water after drip shield degradation or high relative humidity for long periods of time (i.e., thousands of years), the packages will corrode and eventually will be breached. The breaches are expected to occur as deep, narrow pits or cracks, or as broader areas called patches. The changing thermal, hydrologic, and chemical conditions in the potential repository all influence the corrosion rate of the waste packages.

Limiting Radionuclide Mobilization and Release from the Engineered Barrier System–When water eventually enters a waste package through the cracks, patches, or pits, contact may occur with the radioactive waste contained within the waste package. The majority of the radioactive waste is spent nuclear fuel from commercial reactors, but there are also spent nuclear fuel from U.S. Department of Energy reactors, naval fuel, and high-level radioactive waste from the reprocessing of fuel. The commercial spent nuclear fuel is the focus of this overview discussion. The effect on performance of other waste forms are discussed in the main body of the report.

After water enters the waste package, the water will first contact the thin layer (about 0.7 mm) of a zirconium alloy that covers the surface of most of the commercial spent nuclear fuel elements. This layer, called cladding, must be breached by mechanical or chemical processes before the radioactive fuel pellets can be exposed to water. Then the individual fuel elements start to degrade, making the radionuclides (which are distributed in low concentration throughout the uranium oxide fuel pellets) available for transport away from the waste form (Figure ES-7). The degradation process may involve several stages because the waste forms are sometimes altered to different chemical forms (or phases) before they reach a phase that will allow the nuclides (or radionuclide) to be released from the waste into the available water. Also, different radionuclides have different chemical properties themselves, so the reaction rates of the individual nuclides with water are greatly variable. In general, however, once the waste form begins to alter, it takes about 1,000 years for the commercial waste forms to completely degrade.

To move out of the waste package, the radionuclides are either dissolved in or move as extremely small particles (colloids) in flowing water, or they move in a thin stagnant film of water by diffusion. To escape from the waste package, the nuclides must exit through a pit, crack, or patch in the waste package and move out into the waste emplacement drift.

After escaping from the waste package, the radionuclides can then advance through materials on the drift floor, which consists mainly of tuff gravel and the corrosion products from the waste package and drift structural components. At this point, the nuclides may either adhere to some of the materials on the drift floor, continue to move in the water, or become attached to colloidal particles of clay, silica, or iron. Because of their molecular charge and physical size, these colloidal particles move through the rock mass under the potential repository somewhat differently than noncolloidal or dissolved particles.

Slow Radionuclide Transport away from the Engineered Barrier System–The radionuclides move downward beneath the potential repository at different rates based on the chemical characteristics of the nuclides and the rock they are passing through, and on the velocity of the water in which they are contained (Figure ES-8). The rock for several hundred meters underlying the potential repository is unsaturated, and the water movement behaves as described earlier. Some water moves rapidly in fractures and some much more slowly in the rock matrix. Pore water in the matrix also evaporates and recondenses elsewhere due to the ambient geothermal gradient in the mountain; radionuclides left behind in the rock pores by evaporation must dissolve in new water imbibing from fractures, or condensing from vapor, before it can be transported.

The transport rate through fractures and the matrix depends on the tendency of the individual nuclide to interact with the rock through which it moves. Some radionuclides move more quickly through the rock with little or no interaction to delay their transport. Other radionuclides adhere to some minerals in a process called sorption and are bound in the rock for long periods. Sorption can be irreversible in some instances, and in this case the nuclide will be bound permanently in the rock. In other cases, the nuclides may desorb at a future time and again move through the system. Radionuclides also can diffuse from higher concentrations in fracture water to lower concentrations in matrix water, which slows the overall transport rate. It is expected that eventually, some fraction of the available nuclides will travel through the unsaturated zone.

When the radionuclides reach the water table, they will enter the saturated zone flow system. Beneath Yucca Mountain, the water in the saturated zone flows in a generally southerly direction toward the Amargosa Valley. Nuclide sorption also occurs in the rocks and valley fill or alluvium along the flow paths in the saturated zone. Because of the differences in chemistry between the unsaturated and saturated zone rock and water, the rates, durations, and nuclides involved in sorption are different for the two zones. As the radionuclides move in the saturated zone along different paths and through different materials, they gradually become more dispersed and the concentration of the nuclides in any volume of water therefore decreases.

If the radionuclides are eventually pumped out of the saturated zone by water wells, the radioactive material can cause doses to humans in several ways. For example, the water from the well could be used to irrigate crops that are eaten by individuals or livestock, to water stock animals that provide milk or meat food products, or to provide drinking water. Also, if the water pumped from irrigation wells evaporates on the ground surface, the nuclides may be left as fine particulate matter that could be picked up by the wind and then inhaled by humans.

Addressing Effects of Potentially Disruptive Events and Processes–The attributes of the system, given in the previous sections, describe the continually ongoing processes that are expected to occur in and around the potential repository system. The term used to denote the sequence of anticipated conditions is "nominal scenario." In contrast, "disruptive scenarios" refer to discrete, unanticipated events that disrupt the nominal case system (Figure ES-9). Scenarios are developed for this level of analysis. A scenario is a well-defined, connected sequence of features, events, and processes that can be thought of as an outline of a possible future condition of the repository system. The only disruptive event included in this analysis is the formation of a volcano through or adjacent to the potential repository. Other potentially disruptive events were determined not to be significant to overall repository performance and were not included in the TSPA disruptive events analysis. The treatment of these disruptive events in TSPA are discussed in the following paragraphs.

Yucca Mountain's terrain has experienced volcanic activity in the geologic past. The rocks in which the potential repository will be constructed are volcanic in origin. However, scientific studies of the timing, volume, and other aspects of volcanism have concluded that volcanic activity in this area has been waning in the recent geologic past and that the probability of volcanic activity as a potential repository-disturbing event is highly unlikely. Nevertheless, for completeness, part of the total system performance assessment analysis is an assessment of the consequences of a small cinder cone formed by a dike that flowed up through, or close to, the potential repository drifts. Both direct release to the atmosphere of the waste package materials and indirect release to the unsaturated and saturated zones from damaged packages are considered.

Another disruptive event is an earthquake, or seismic activity. Although generally modest in size, earthquakes do happen frequently in and around Yucca Mountain. The seismic hazard exposure of the potential repository primarily results from ground motions rather than from direct offset along a fault. The primary potential effect of ground shaking is to disrupt the cladding of the waste and hasten rockfall into the drift. The effects of rockfall and seismic effects on the waste form are included in the nominal case analysis.

In previous total system performance assessment calculations, the effects of nuclear criticality, another potential disruptive event, have been assessed for both in-waste package and in-rock events. In those analyses, a series of unlikely events was assumed to occur. These unlikely events (such as filling the waste package with water or concentrating specific radionuclides in the rock mass) lead to the concentration of certain nuclides that, only in specific low-probability environments, might lead to a nuclear criticality. The result is a change in the nuclear material to more highly radioactive forms. The resulting increase in the radionuclide source term was then evaluated against the base case to determine if the resulting change in dose rate is significant, and it is not. Because the probability of occurrence of an in-package or a rock mass criticality is very low and the consequence of a criticality (should one occur) on the radiation dose is also very low, further analyses are not presented here.

Human Intrusion–Another disruptive event, human intrusion into the potential repository, is treated as a separate scenario in the analysis. Human intrusion is treated in a stylized manner based on the proposed regulatory description of such an event in which the contents of a waste package are exposed through the borehole of a well drilled directly through the potential repository into the uppermost aquifer 100 years after closure. The human intrusion is assumed to occur and the dose consequences are calculated and compared against the nominal scenario results to evaluate the robustness of the potential repository in the event of such an intrusion.

Results of Analyses–Although the total system performance assessment is usually discussed in terms of a sequence of processes linked one after the other in space (as described in the earlier section), this approach does not readily convey how all of the processes evolve with time. The following describes the results at various time intervals of interest, attempting to show the evolution in both time and space for the reference design and for the range of nominal case conditions. However, the assumptions underlying the modeling development drive the results. Different sets of assumptions can give different results. The intent of this total system performance assessment is not only to show how the system is thought to behave, but also to provide information on how much uncertainty is associated with each total system performance assessment component, as discussed later. Many of the results shown include a great deal of conservatism and also some large ranges of uncertainty. Conservatism is utilized in the analyses to add defensibility to the analyses in the case where a parameter or model has an uncertain range of performance. The conservative approach tends to promote under performance of the component in question. In terms of evaluating the safety of the site in a regulatory framework, this is a better approach to take than to perhaps trend toward less defensible, over performance of the potential repository because it may require less resources to defend the analyses. The results discussed below focus on the forecasted time-averaged behavior of the repository system. This behavior by itself cannot fully represent the ranges of uncertainty and variability in the system and its possible future states.

The approach to performance assessment model development is shown in Figure ES-10. The figure illustrates the identification and screening of the features, events, and processes, followed by modeling of various components of the potential repository system for each of the main scenarios.

How the Potential Yucca Mountain Repository is Projected to Evolve–Prior to describing the probable evolution of the repository system it is worthwhile to note that although the illustrations depict how the system is projected to degrade over time, there are large parts of the system that remain essentially unaltered for very long periods. Although this fact is reflected indirectly in the results, it is rarely shown explicitly. The sequence of results in Figures ES-11 to ES-15 show schematically what the waste package and engineered system might look like at various times after closure. However, the schematics are only representative of those packages that experience dripping water (or seeps). The percentage of all waste packages that experience significant corrosion of the resistant outer high-nickel alloy layer is expected to be small until late times (i.e., several tens of thousands of years). Even though the location and number of seeps changes with time, the majority of waste packages will likely never experience any significant seepage, even in a million years. Most of the waste packages are expected to remain relatively intact and continue to look essentially like the one depicted in Figure ES-11 with only minor breaches and not like the waste packages shown in subsequent figures depicting the total system performance assessment results. Note however, that waste packages may degrade even in the absence of seepage, due to moisture on the waste packages contributing to degradation.

Waste Emplacement to Several Thousand Years after Potential Repository Closure–As the waste packages are emplaced in the drifts, their combined heat output will cause the drift wall temperatures to rise, and much of the water and gas in the rock will be driven away from the potential repository. At 100 to 200 years after closure, the surfaces of some of the individual waste packages will start to cool below boiling, and the humidity in the drift will climb from preclosure values of about 50 percent to nearly 100 percent. Depending upon the local conditions around each waste package, the degradation of the high-nickel alloy outer layer will begin somewhere between 100 years and several thousand years, but proceed at a very slow pace. As a result, waste packages are not expected to be breached such that radionuclides could be exposed to water, if for several thousand years.

In Figure ES-11, the cutaway of the potential repository shows schematically the situation under nominal conditions at 1,000 years. There is not any forecasted release from the potential repository system at this time. Therefore, as shown in the final panel of Figure ES-11, there is no dose consequence for any of the cases calculated during this period in the region 20 km downgradient of the potential repository. The schematic picture of the waste package in Figure ES-11 shows an intact waste package that exhibits little or no corrosion or degradation. Note that the natural environment also provides an independent barrier for radionuclide transport for the first 1,000 years. This means that the bulk of the radionuclide inventory in the system will not be released to the accessible environment, either due to the engineered system or the natural system, and will not contribute to the expected dose to the average member of the critical group.

Several Thousand Years to 10,000 Years after Closure–The expected value of the peak dose is a function of the degree of conservatism incorporated in the models and analyses used to produce the peak dose estimate. Because the base case models used in the development of the nominal performance projections were designed to be reasonably conservative to maximize their defensibility during the 10,000-year compliance period, they are less appropriate for projections of the peak dose. More appropriate representations would include considerations of the long-term (post-10,000-year) climate states and the long term effects of secondary phases. The following discussion relates primarily to the nominal case for 10,000 year model. For the nominal scenario, the potential repository system still performs very well during this time period. The waste packages show minor corrosion. The heat in the system has begun to decay, and reduce back to ambient conditions. The large majority of the radionuclide inventory itself has decayed by this time. Figure ES-12 shows the lack of corrosion and release from the potential repository system. Note that the proposed regulatory compliance period is only for 10,000 years. Discussion of time periods past the 10,000 year compliance period are intended only to provide a context for better understanding of the compliance-period results. After a few thousand years, the engineered system is utilized to isolate the remaining small percentage of the inventory that has not decayed away at this time.

10,000 Years to 50,000 Years after Closure–By this time, the rock surrounding the drift is returning to its original temperature, and the original fluid flow patterns affected by the heating have been reestablished. Some permanent alterations of the rock may remain (such as fracture shear movement caused by thermal expansion and contraction), but this does not appear to be significant in terms of potential repository performance. The outer layer of the waste package continues to corrode, though very slowly. Dripping conditions now occur at discrete locations throughout the potential repository. Where the outer layer or a weld on the lid of the waste package has been perforated, corrosion of the inner barrier material is initiated (Figure ES-13). Inner barrier corrosion proceeds much more quickly than that for the outer layer. In the cases where the inner layer has been perforated, the water can enter the waste packages through small openings, alter the fuel in rods that have been perforated, and move out of the engineered barrier system. The potential repository cutaway in Figure ES-13 shows a few paths along which nuclides are being released into the rock under the potential repository. At this time, the median value for the number of breached waste packages is less than 10 percent of the total emplaced packages. The mean peak dose rate from a plume in the saturated zone 20 km south-southeast of the potential repository is calculated to be 0.25 mrem/year, primarily from ^{99}Tc and ^{129}I. This value is 0.08 percent of the average background radiation from nonmedical sources in the United States, which is about 300 mrem/year. Background non-medical radiation in the U.S. varies with location. For example, it is about 310 mrem/yr in Oak Ridge, Tennessee, 340 mrem/yr in Amargosa Valley, Nevada, and 1180-mrem/yr in Denver, Colorado. (see DOE 1999 [105155], Table 3-28, Volume 1). Radon dose contributes a substantial fraction of the background dose rate, about 200 mrem/yr on average. The radon dose around the country varies, with higher dose for uranium-bearing underlying rock, the use of basements in building construction, and tightly-sealed, energy-conserving buildings. The values listed above are based on an average radon exposure, except for Denver, which uses a location-specific value.

Another way to assess the ability of the potential repository system to isolate the waste is to show the radionuclide release information in the context of how much radioactivity remains isolated in the potential repository versus how much has escaped. Figure ES-16 illustrates how the total inventory in the potential repository at the time of emplacement (30 years) decays with time, and what amount of the total inventory has escaped from the potential repository at discrete times up to 100,000 years. Compared to the total amount of radioactivity in the potential repository at 30 years after waste emplacement at about 300 years after closure, the decay process has decreased the radioactivity to about 2 percent of the original amount. At 1,000 years the amount has decreased further to 0.8 percent, and at 10,000 years the remaining portion of the original total inventory is only about 0.2 percent. Of the 0.2 percent remaining at 10,000 years, none is projected to reach the edge of the potential repository. At 100,000 years, 0.01 percent of the original inventory remains. Of that remaining 0.01 percent, 3 percent is projected to reach the edge of the potential repository, 2 percent to reach the water table and be transported 20 km south of the edge of the potential repository, where it is assumed to be accessible to humans.

Fifty Thousand to 100,000 Years after Closure–The natural conditions in the rock remain unchanged from the previous period. The progression of corrosion of the packages is shown in Figure ES-14. Those nuclides that at earlier times are limited in their release from the spent nuclear fuel elements because of their chemistry become larger contributors to the dose rate. In particular, ^{237}Np becomes the dominant isotope controlling dose rate. The median number of packages breaching by the end of this time is about 50 percent of the total number. The mean peak dose rate at 20 km is 70 mrem/yr, or about 23 percent of the average background radiation from natural sources in the United States.

One Hundred Thousand to 1 Million Years after Closure–The individual waste packages continue to slowly corrode. The number of packages releasing nuclides by 1 million years after closure is about 100 percent of the total (Figure ES-15). Dose rates at the 20 km point continue to climb as more packages release their inventory, until a maximum is reached at approximately 250,000 years. The mean value for total dose rate at this time is approximately 460 mrem/yr and then declines to approximately 180 mrem/yr at 1 million years after closure. Although 400 mrem/yr due to repository pathways is about double the present nonmedical background dose rate in the Yucca Mountain area, it is well within the natural variability of background dose rates in the United States. Residents of Denver, Colorado for example, receive about triple the present background dose rates in the Yucca Mountain region. ^{237}Np remains the main contributor to the dose rate, but plutonium attached to colloids is the dominant contributor in some of the cases.

As noted in proposed 40 CFR 197.30, (64 FR 46976 [105065]), no regulatory standard applies to the results of the peak dose analyses. They are provided to support the development of the environmental impact statement EIS. Although these results do provide insights into the possible long term performance of a repository at Yucca Mountain, they should not be interpreted as accurate predictions of the likely performance over these time periods due to the large uncertainties and conservative approximations included in the models that were designed for assessing the 10,000-year compliance performance.

Sensitivity Analysis–An important role of performance assessment is to evaluate the uncertainty in the projected performance and the significance of the key component models and parameters in the projection. In general, the sensitivity analyses show, in a relative way, the parameters in which uncertainty most affects the results. In some cases, if future studies could reduce the range in uncertainty, the parameter might no longer appear as a parameter to which performance is highly sensitive. Conversely, if a parameter or component is assigned an inappropriately low uncertainty range, it might not show up as a particularly important parameter. These analyses must be performed with care to gain the necessary understanding about the parameters that are most important to actual repository performance.

Based on the sensitivity and uncertainty analyses of the total system performance assessment results, the following aspects of the total system performance assessment components have been determined to be most significant to the dose rates at 20 km from the potential Yucca Mountain repository. In some cases, the total system performance assessment results point to very specific aspects or parameters used to represent the total system performance assessment components, which in turn are captured in the attributes of potential repository performance. The results are shown for four different time periods because the relative importance of different aspects of the modeled system changes as the system evolves. The results are ranked from most important to least within each time period.

Table ES-1. Important Components of Potential Repository System for Different Time Periods

Performance Period	Most Sensitive Components or Parameters
Postclosure to 10,000 Years	Occurrence of volcanic event disrupting waste packages
10,000 to 50,000 Years after Closure	Occurrence of volcanic event disrupting waste packages
	Availability of water to contact the waste package (seepage into drifts)
	Rate of waste package degradation (loss of integrity of outer waste package barrier or of inner waste package barrier due to environmental conditions)
	Rate of cladding degradation (integrity of spent nuclear fuel cladding)
	Availability of water to contact exposed waste form surfaces (water into waste package)
Fifty Thousand to 100,000 Years after Closure	Rate of waste package degradation
	Rate of cladding degradation (integrity of spent nuclear fuel cladding)
	Neptunium Solubility
	Formation and transport of radionuclide-bearing colloids
One Hundred Thousand to 1 Million Years after Closure	Rate of waste package degradation
	Cumulative amount of degraded cladding (integrity of spent nuclear fuel cladding)
	Neptunium solubility

The details of the total system performance assessment parameters that feed these factors are described in several sections of this document. A brief, general summary of the main analyses is presented below.

Disruptive Events–Several disruptive, or unexpected, features, events, and processes are included in the analyses. The primary disruptive event in TSPA-SR is the volcanic scenario. This case is simulated and is the only contributor to the dose at times before 10,000 years, the proposed regulatory period. The mean peak dose is significantly below the standard during the first 10,000 years of the simulations. The analyses were conducted out to later times as well.

Human Intrusion–A disruptive event defined or stylized in the proposed regulation, is human intrusion. The nominal scenario for the TSPA-SR was utilized, and a stylized human intrusion was included. The case was run probabilistically for a 100,000 year time period, with stochastic parameters for many aspects of the case. The case included a borehole through a relatively intact waste package at 100 years after closure, that penetrated all the way to the saturated zone. The major components of the model include infiltration of water down the borehole into the penetrated waste package; mobilization and release of the waste within the package; transport of radionuclides down the borehole to the water table; transport of the radionuclides through the saturated zone; and biosphere exposure pathways as in the nominal case. Many aspects of this model are uncertain, and were varied in the probabilistic case. The doses generated from this case were significantly below the nominal and disruptive cases at late times, though there were early releases due to the penetration of the waste package at 100 years. A sensitivity analysis was conducted to look at a more realistic time of penetration of the waste package with a drill bit. The later intrusion time was based on degradation or thinning of the waste package, and also resulted in doses well below that of the nominal and disruptive cases.

Groundwater Protection Case–An additional case was also evaluated for TSPA-SR that incorporated pertinent radionuclides and simulated the concentration of radionuclides in the groundwater. The analyses indicate that the groundwater concentrations will be low, even beyond the proposed regulatory time period of 10,000 years.

Summary–The general total system performance assessment process has developed over time through its application on numerous projects by various international organizations involved in radioactive waste management and in consultation with the U.S. Nuclear Regulatory Commission. The TSPA-SR is the fifth major iteration of TSPA conducted by the DOE over the past decade in support of evaluating the suitability of the Yucca Mountain Site. It is based on internationally accepted approaches, including an initial development of features, events, and processes that may occur at the site. Individual models are based on appropriate site-specific information, analog data and relevant literature data sources that have been integrated by the principal scientific investigators to provide a reasonable and defensible characterization of each individual process relevant to postclosure performance The data, analyses, and models used as the technical basis for the TSPA-SR, as well as the assumptions, uncertainty, variability and conservatism that go along with these data, analyses and models are all traceable back to their source documents and data sets. This traceability allows all interested reviewers to examine the defensibility of the individual component models.

The current TSPA-SR Rev 00 has benefited from reviews of the TSPA-VA completed by a Peer Review Panel (Budnitz et al. 1999 [102726]), the NRC (Paperiello 1999 [146561]), Clark County, NV (Cohen 1999 [151783]), and the U.S. Geological Survey (Anderson et al. 1998 [101656]). Section 6.2 presents a summary of many of the most significant comments and how they have been addressed. There remains uncertainty in the individual process models and their abstraction into the TSPA-SR model. Much of this uncertainty has been quantified and is included in the TSPA-SR model. The TSPA-SR results reflect this quantified uncertainty. In addition to the quantified uncertainty in the TSPA-SR model, there is also unquantified uncertainty that has been generally represented by using a more bounded or conservative representation of a particular process model.

All of the above information and their integration in the context of this TSPA-SR provide a sound, traceable, and transparent technical picture of the possible performance of a potential repository at Yucca Mountain. These projections have incorporated the best available science and technology developed over years of investigating the Yucca Mountain site and the associated waste forms and waste packages. Although significant uncertainty exists in some of the component models underlying the TSPA-SR, these uncertainties have either been reasonably quantified, or in some cases of great complexity, conservatively bounded.

The documentation of the TSPA-SR, including the analysis model reports, process model reports, and the TSPA-SR model document, provide the scientific basis for evaluating the suitability of the Yucca Mountain site and for addressing the NRC acceptance criteria in the Total System Performance Assessment and Integration Issue Resolution Status Report.

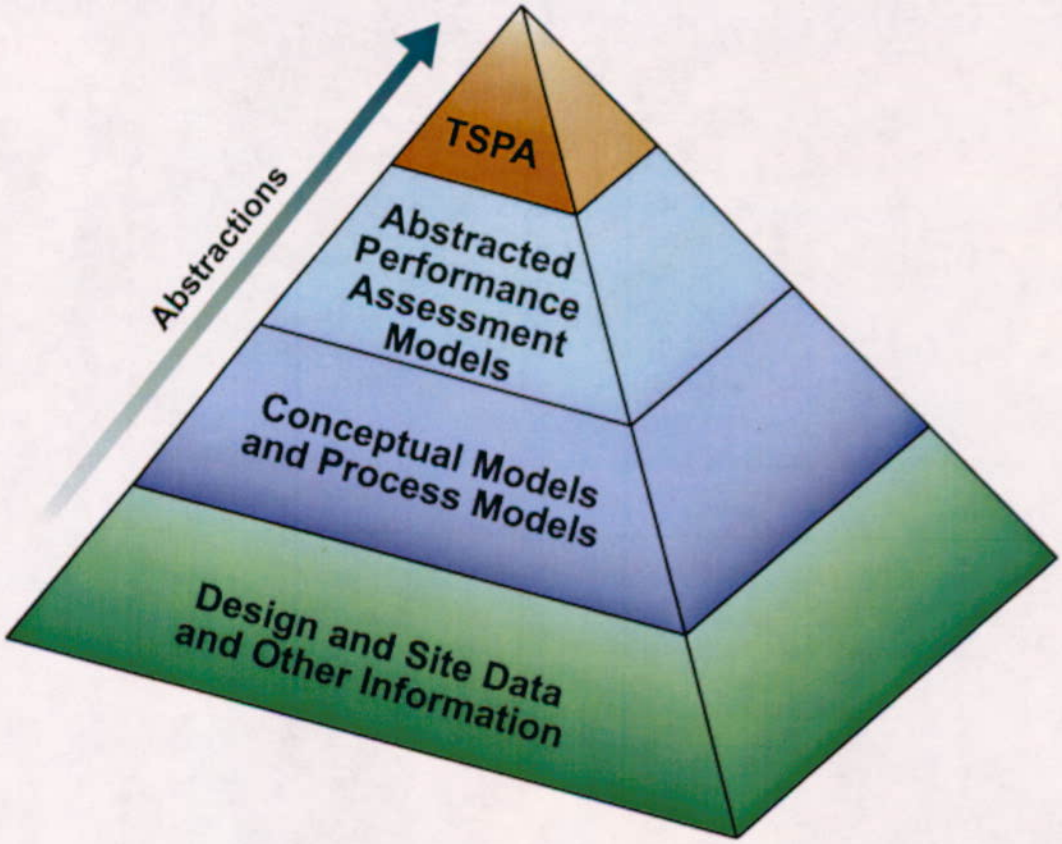

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Figure ES-1. Total System Performance Assessment Information Pyramid

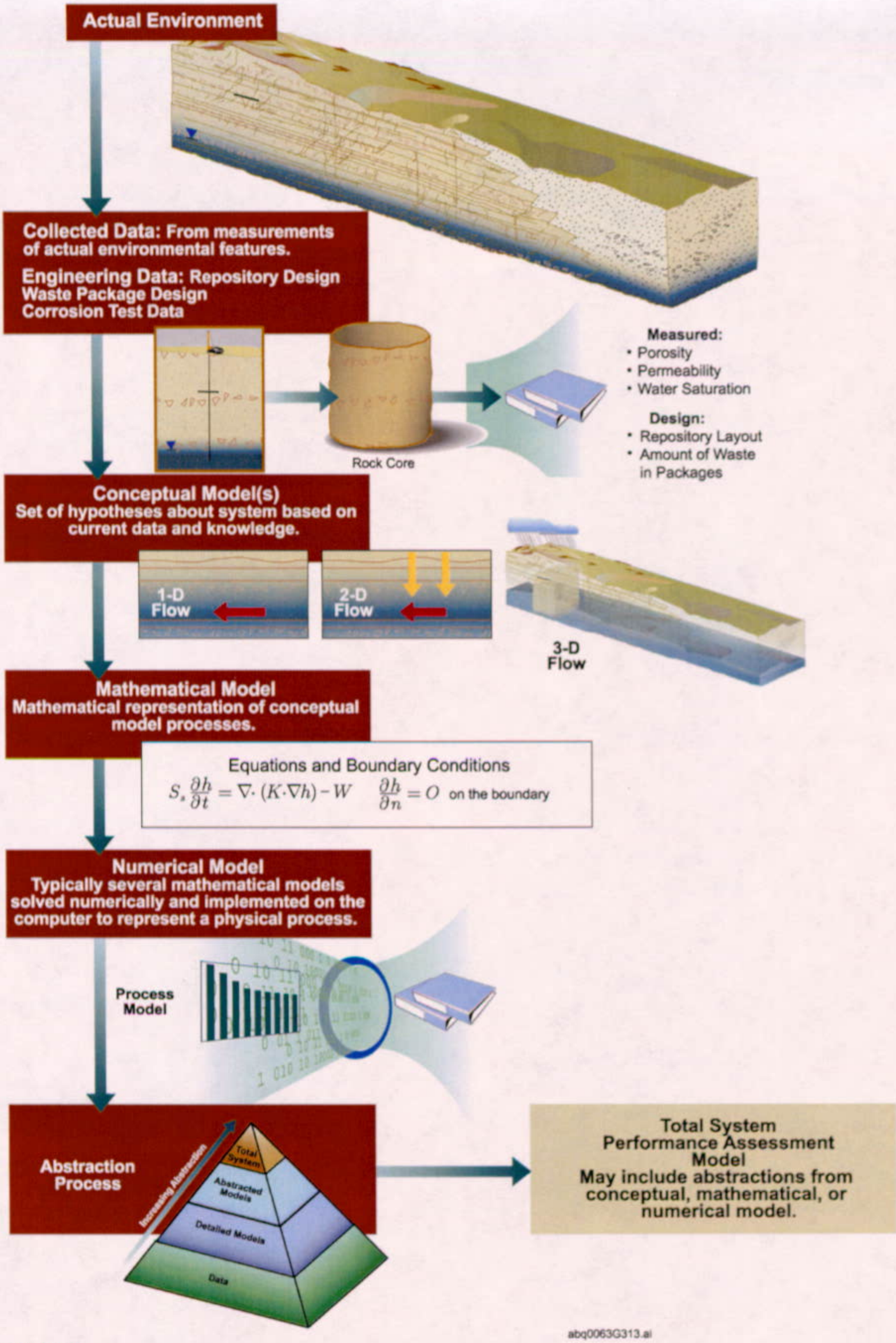

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Figure ES-2. Generalized Performance Assessment Approach

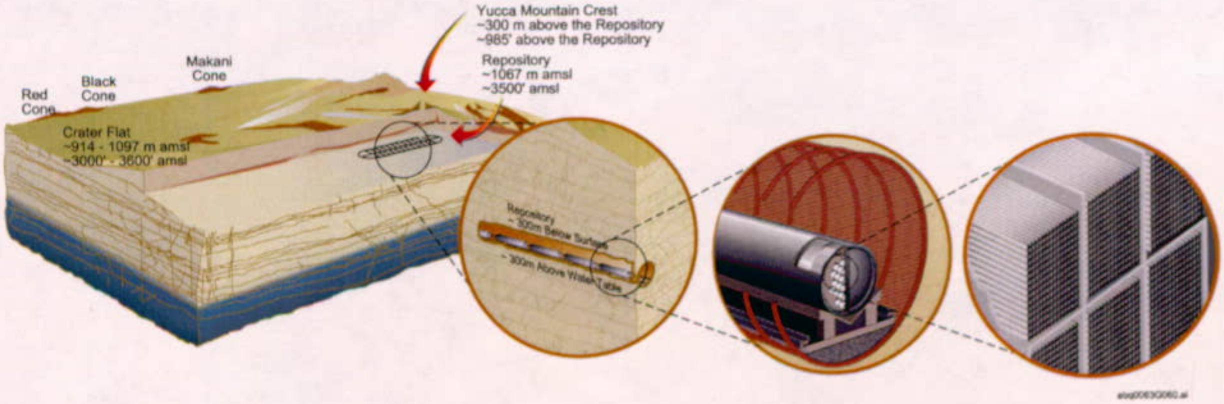

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Figure ES-3. Generalized Schematic of Potential Repository System from Mountain Scale to Repository Scale to Waste Package Scale to Waste Form Scale

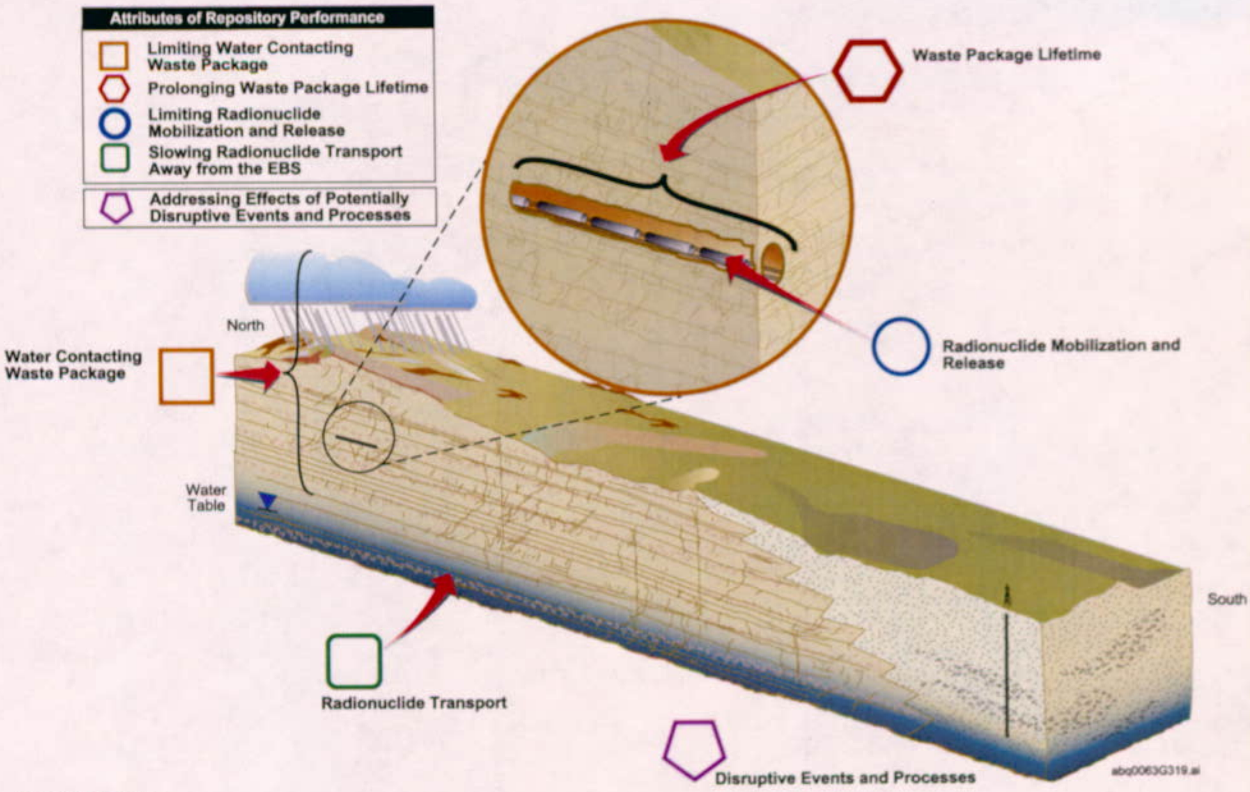

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Figure ES-4. Schematic of Attributes of Repository Performance

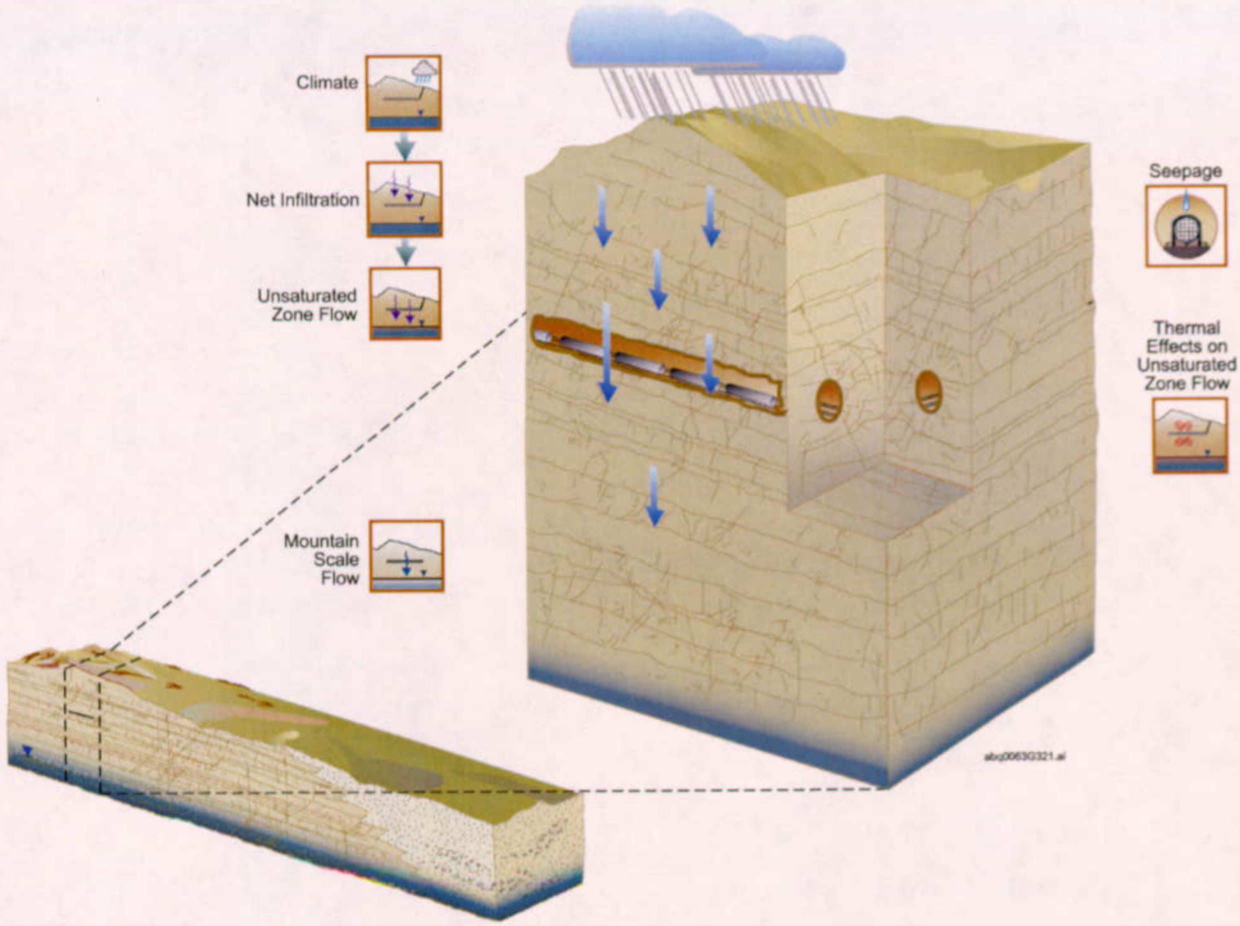

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Figure ES-5. Limiting Water Contacting Waste Package Attribute

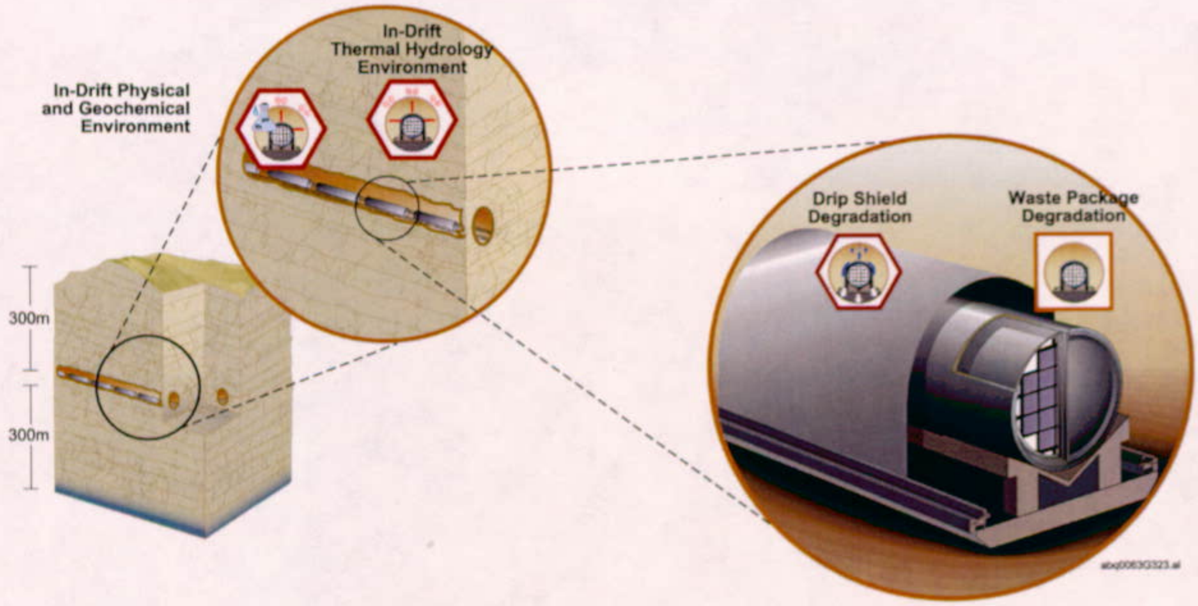

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Figure ES-6. Prolonging Waste Package Lifetime Attribute

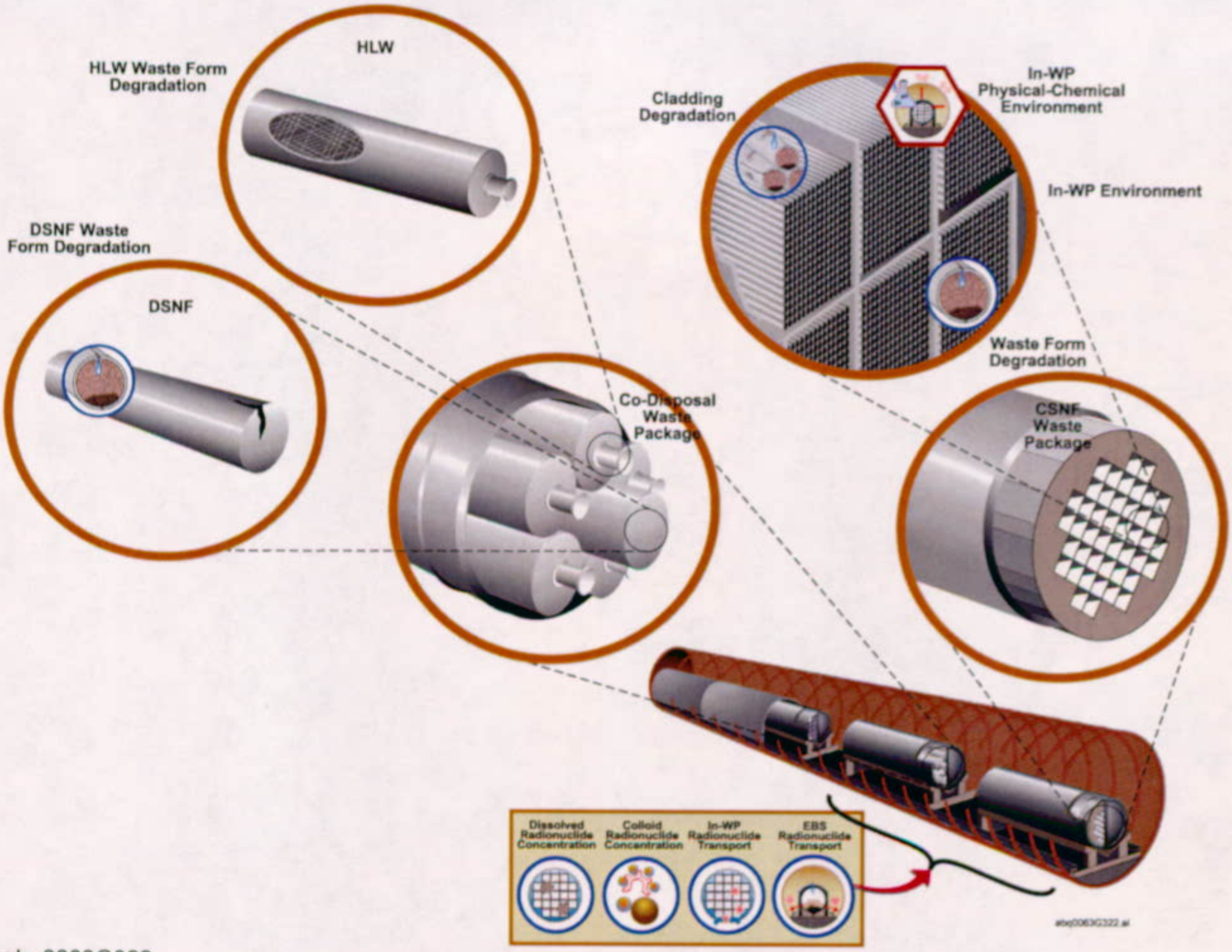

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Figure ES-7. Limiting Radionuclide Mobilization and Release Attribute

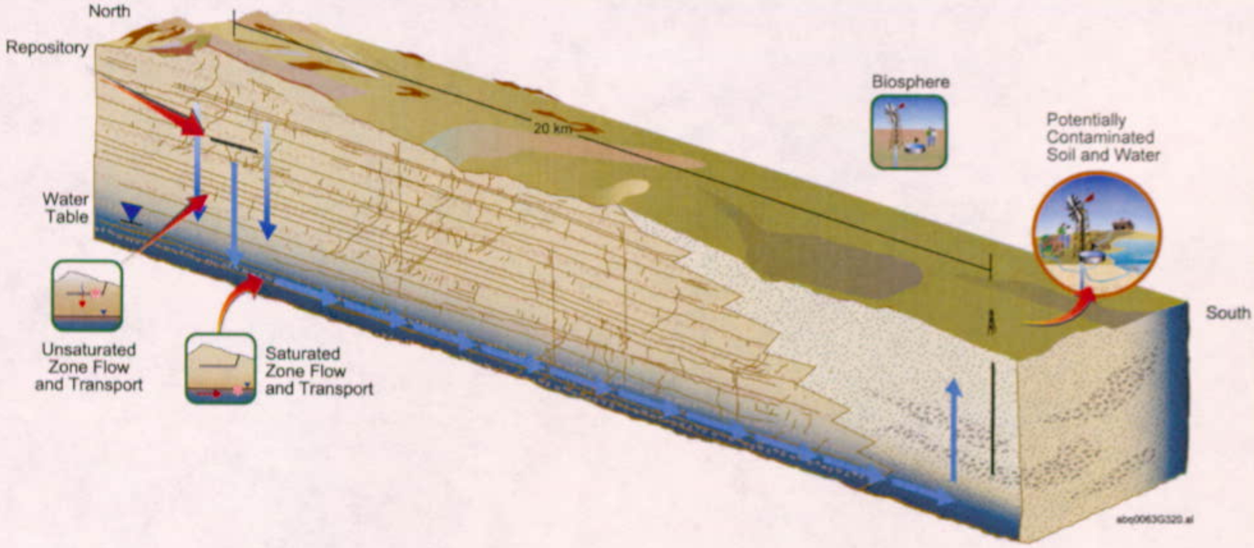

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Figure ES-8. Slow Radionuclide Transport Away from the Engineered Barrier System Attribute

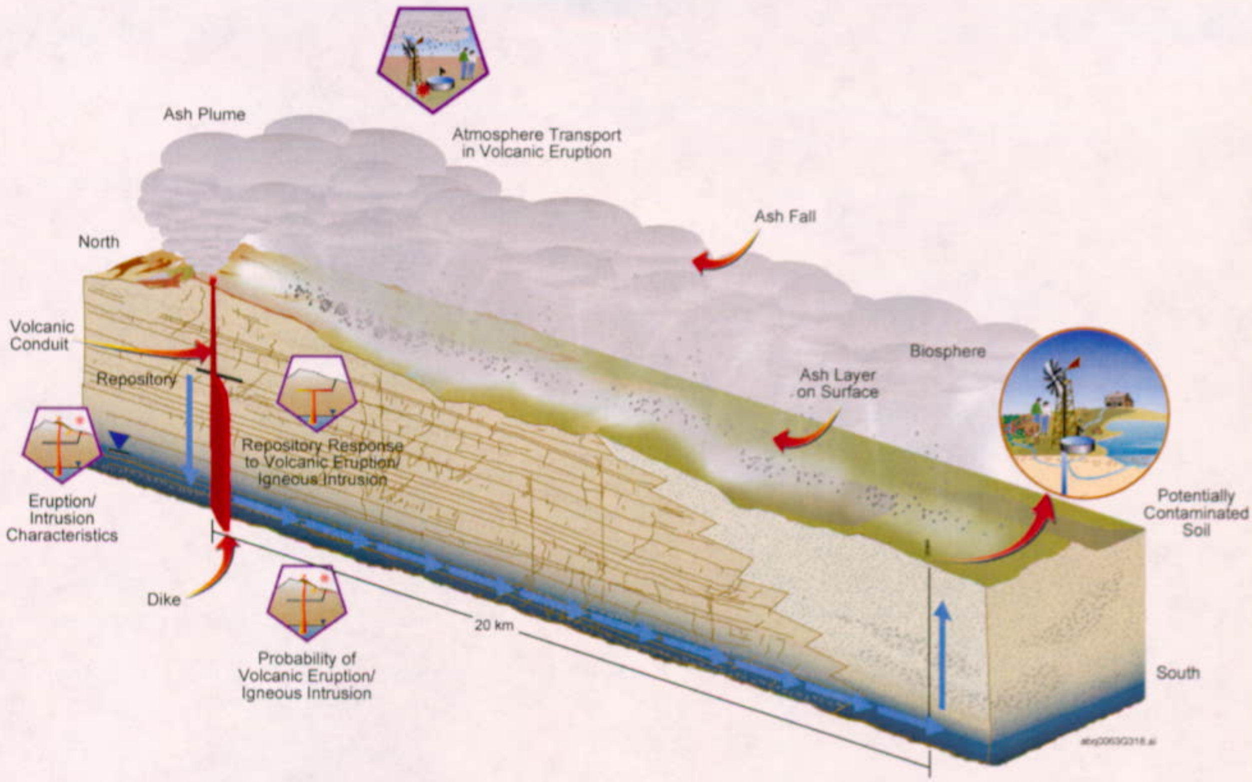

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Figure ES-9. Addressing Effects of Potentially Disruptive Events Attribute

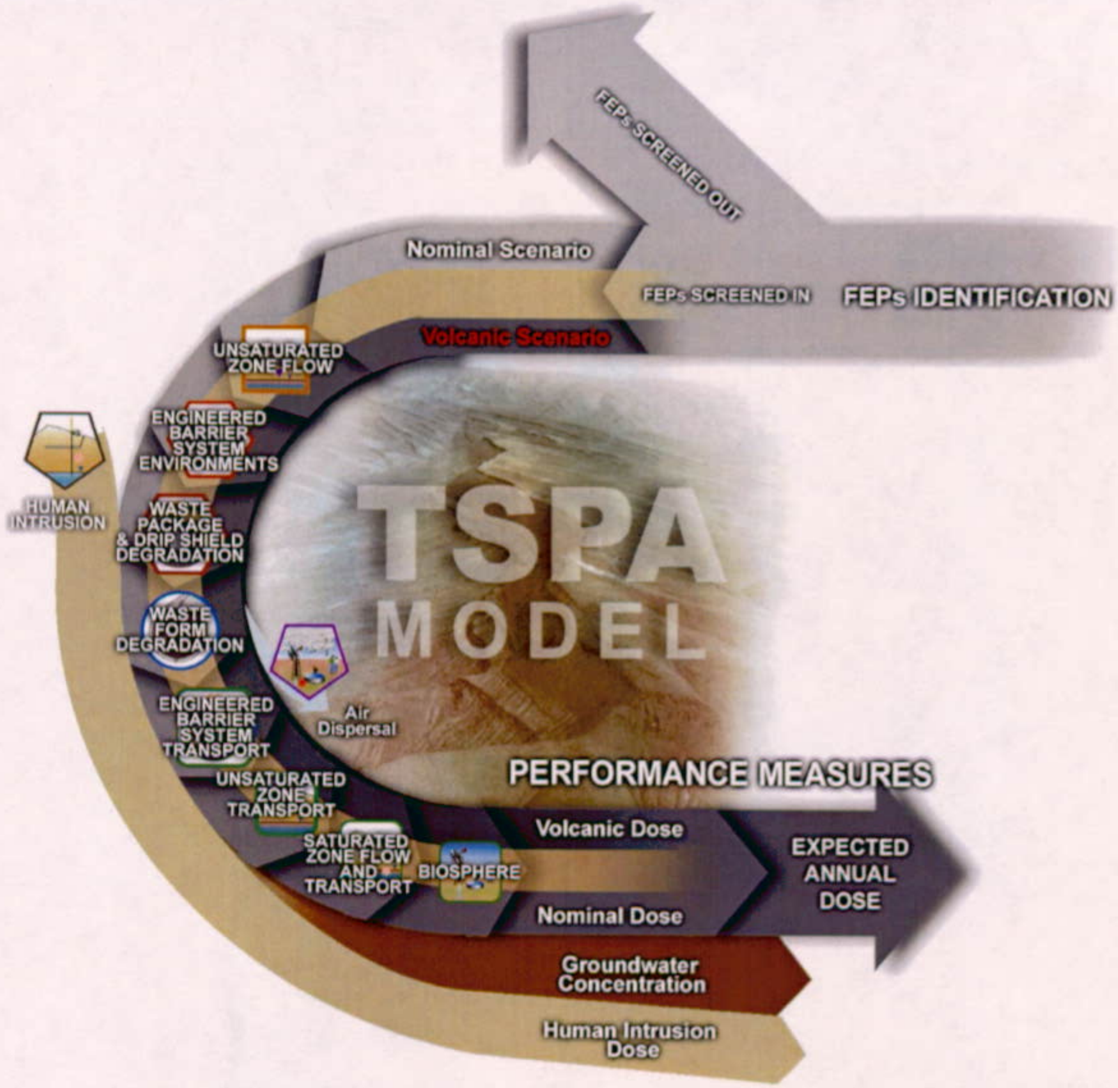

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Figure ES-10. Schematic Representation of the Development of TSPA-SR including the Nominal, Disruptive, and Human Intrusion Scenarios

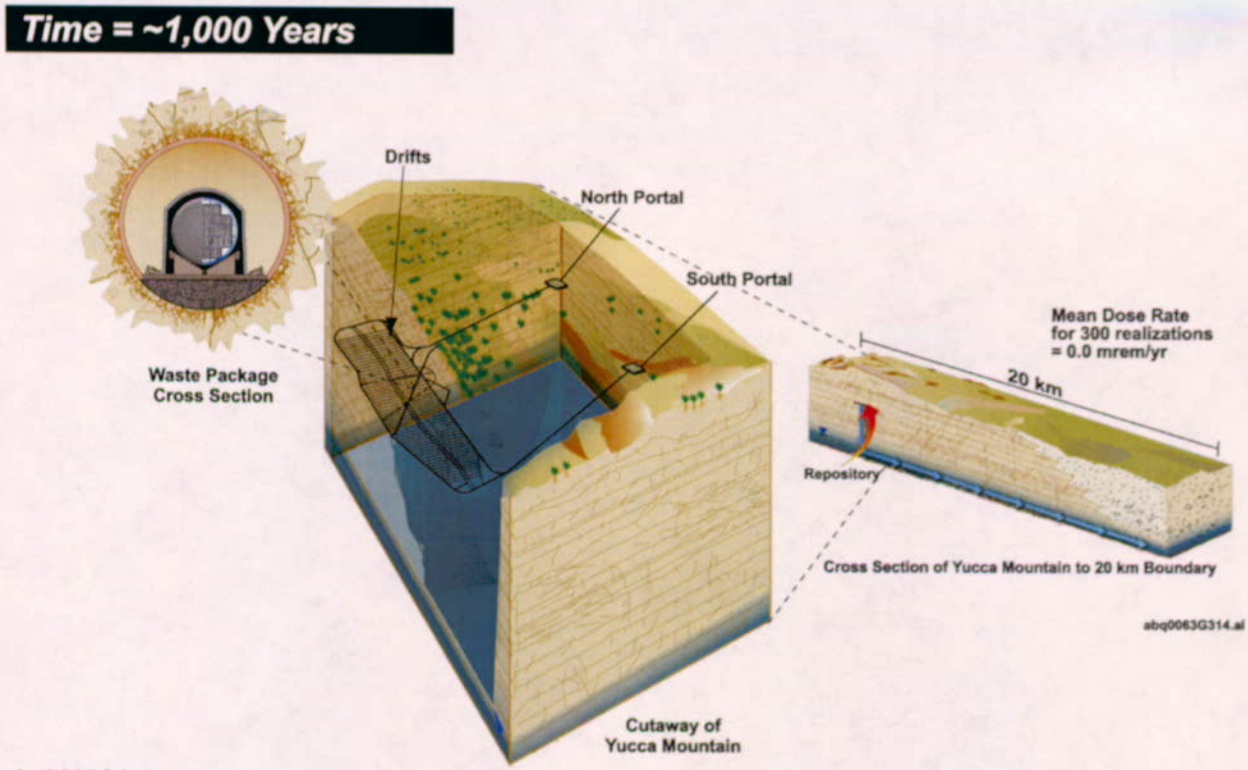

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Figure ES-11. Potential Radionuclide Release Conditions at About 1,000 Years

Time = ~10,000 Years

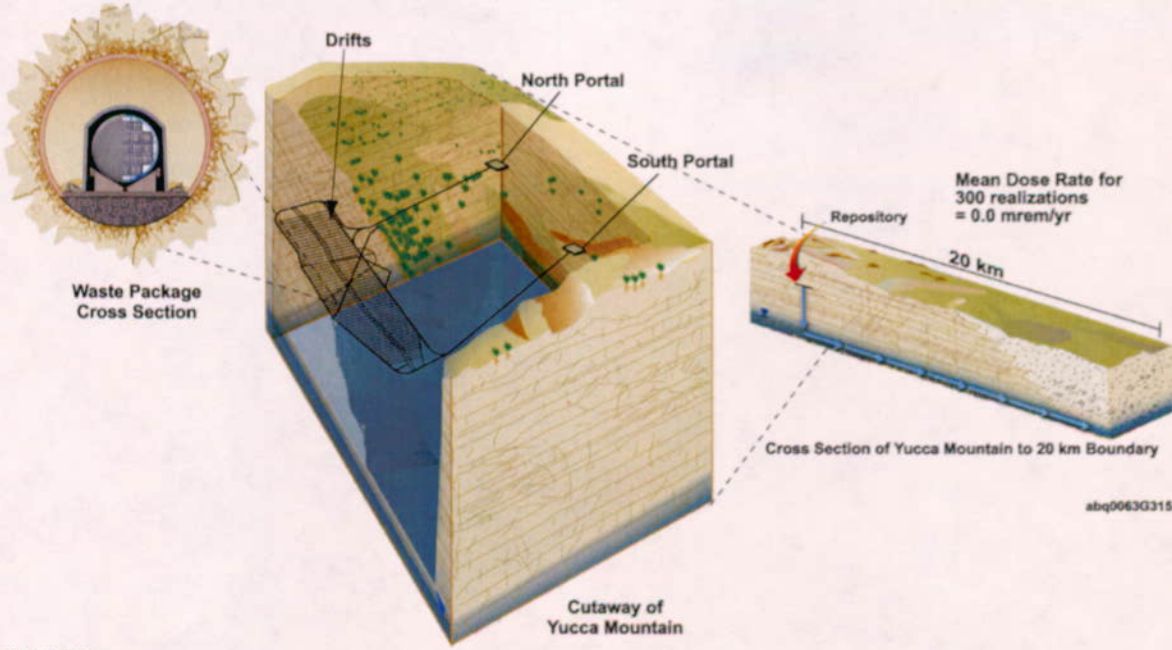

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Figure ES-12. Potential Radionuclide Release Conditions at About 10,000 Years

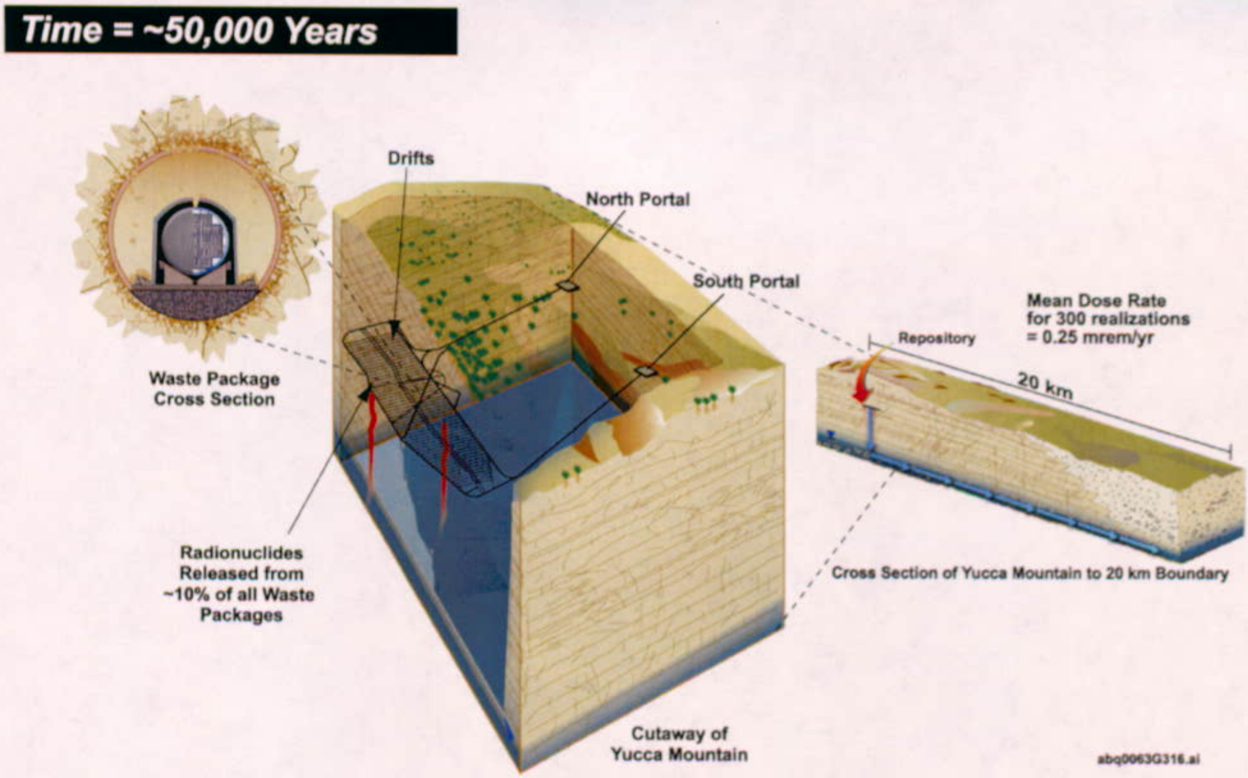

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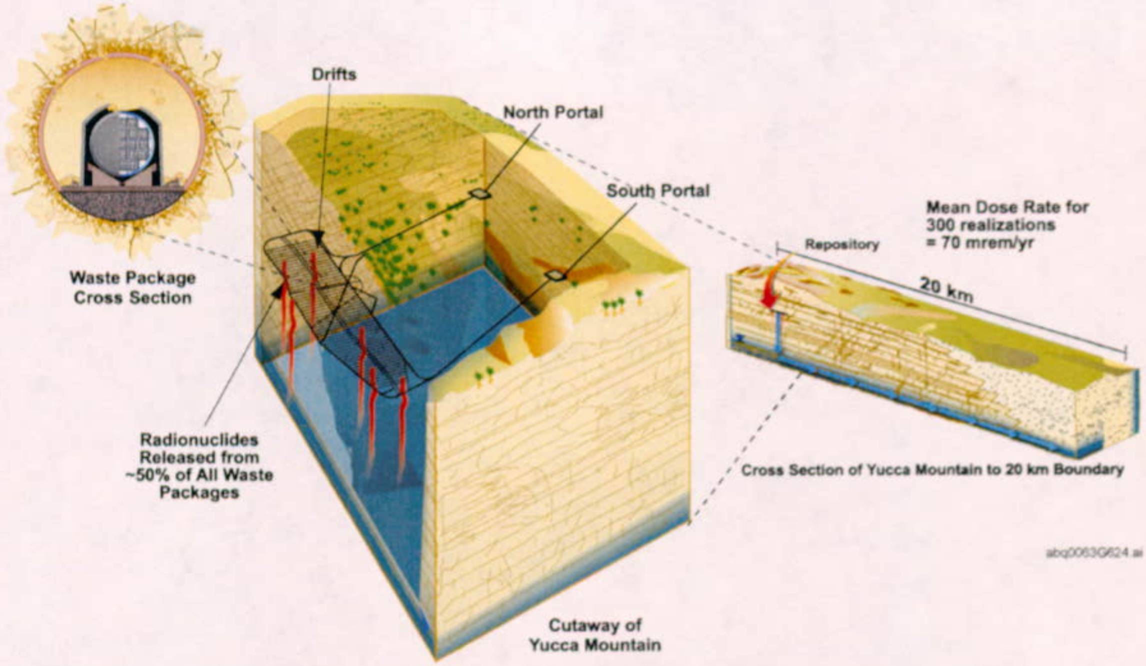
Figure ES-13. Potential Radionuclide Release Conditions at About 50,000 Years

Time = ~100,000 Years



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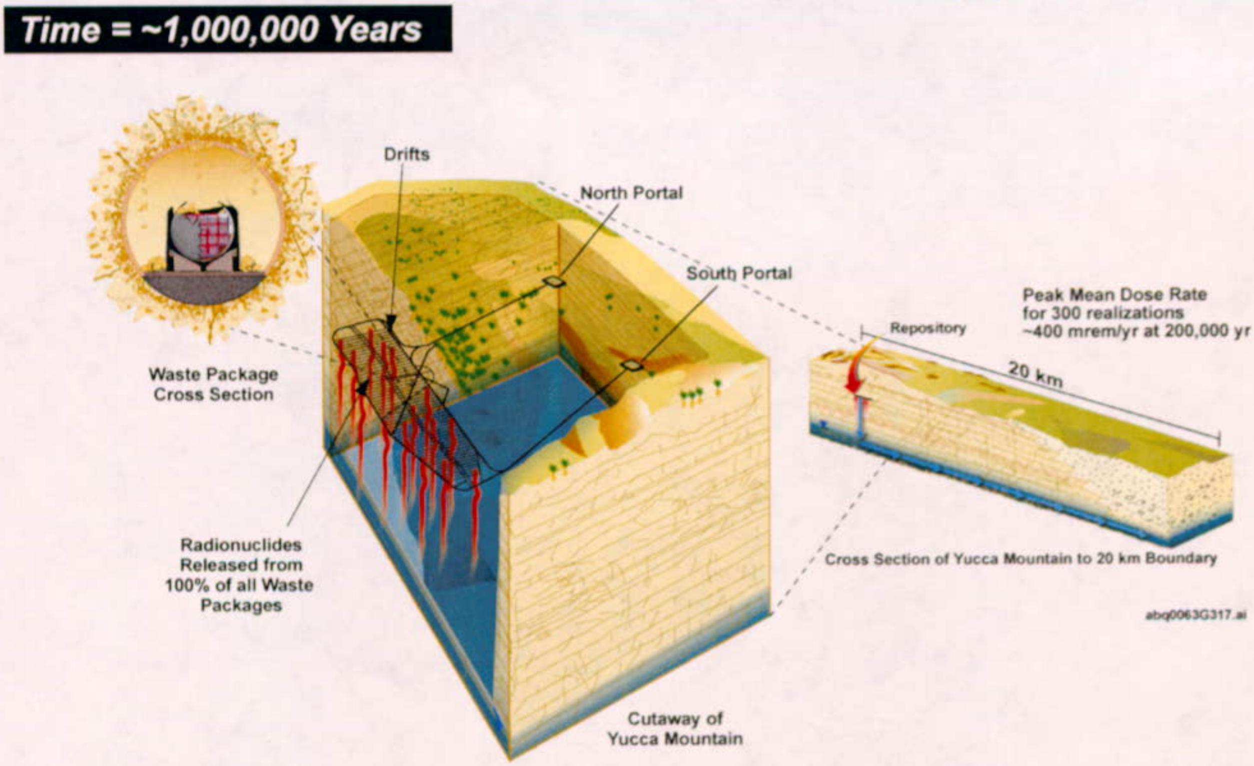
NOTE: The waste package cross section in the upper left is only applicable to the portion of waste packages experiencing the environmental conditions that may cause extensive general corrosion. The TSPA analyses detail the small number of waste packages that will experience such failure.

Figure ES-14. Potential Radionuclide Release Conditions at About 100,000 Years

![Time = ~1,000,000 Years]



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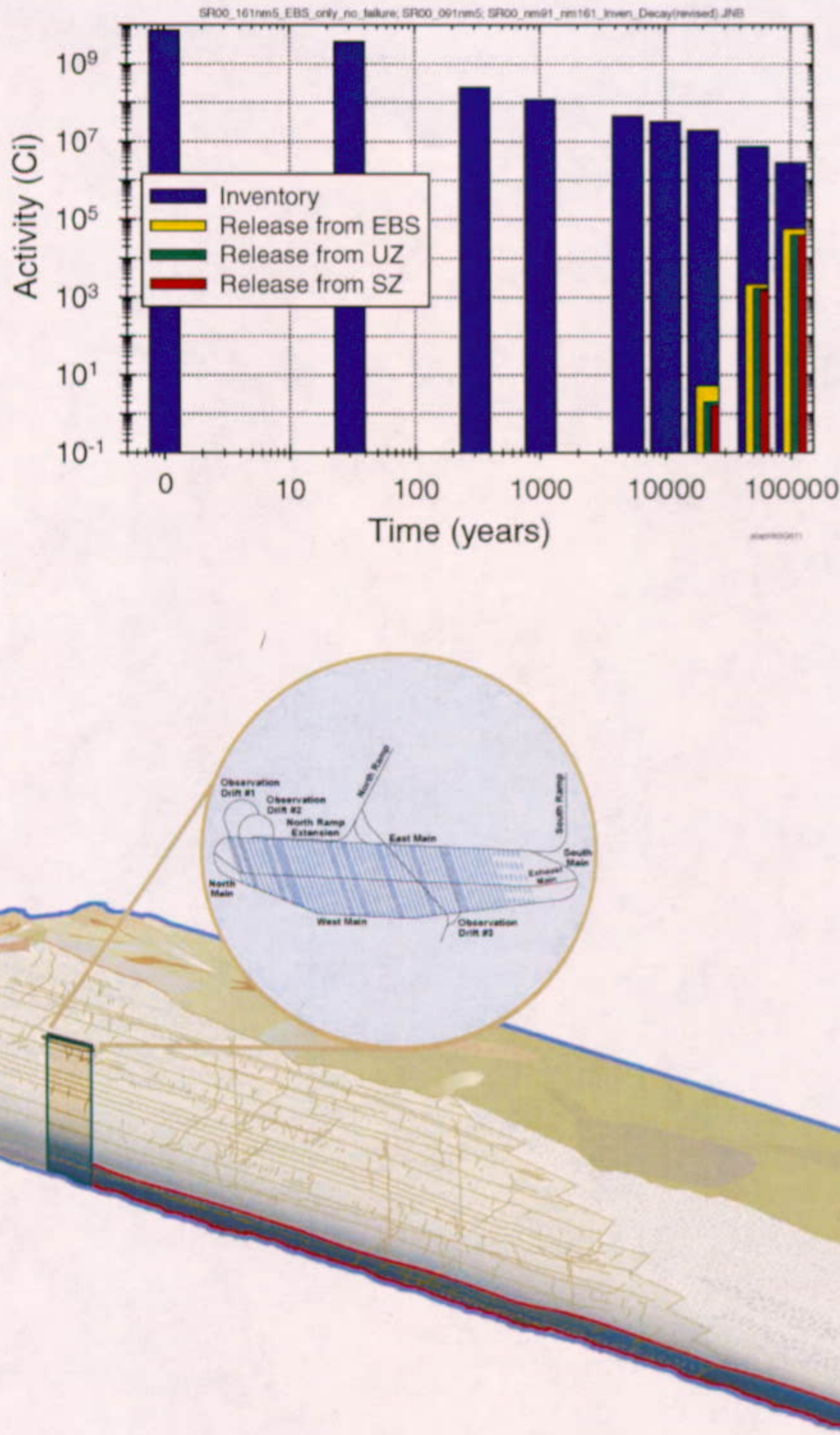
NOTE: The waste package cross section in the upper left is only applicable to the portion of waste packages experiencing the environmental conditions that may cause extensive general corrosion. Section 3.4 details the small number of waste packages that will experience such failure.

Figure ES-15. Potential Radionuclide Release Conditions at About 1,000,000 Years



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NOTE: The blue portion of the bar shows how much of the total radioactivity (in Curies) remains in the system at several points in time. The yellow portion shows the percentage of the remaining radioactivity that has passed through the floor of the repository into the underlying rock, the green indicates the amount of radioactivity that has reached the water table, and the red shows the amount that has moved to the accessible environment 20 km (12 miles) south of the repository. EBS - Engineered Barrier System; UZ - Unsaturated Zone; SZ - Saturated Zone.

Figure ES-16. Progressive Loss of Radioactivity Due to the Decay Process

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ACRONYMS AND ABBREVIATIONS

AMR	Analysis Model Report
BDCF	biosphere dose conversion factor
BWR	boiling water reactor
CDF	cumulative distribution function
CHLW	commercial high-level radioactive waste
CRM	corrosion resistant material
CSNF	commercial spent nuclear fuel
DFEP	disruptive feature, event, and process
DIRS	Document Input Reference System
DOE	U.S. Department of Energy
DSNF	DOE-owned spent nuclear fuel
EBS	engineered barrier system
EFEP	expected feature, event, and process
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
ESF	Exploratory Studies Facility
FEP	feature, event, and process
GVP	Gaussian variance partioning
HIC	hydrogen induced cracking
HLW	high-level radioactive waste
IRSR	Issue Resolution Status Report
KTI	key technical issue
LA	license application
MIC	microbiologically influenced corrosion
MTHM	metric tons of heavy metal
MTU	metric tons of uranium
NAS	National Academy of Sciences
NEA	Nuclear Energy Agency
NRC	U.S. Nuclear Regulatory Commission
NTS	Nevada Test Site
NWTRB	U.S. Nuclear Waste Technical Review Board
PMR	Process Model Report
PWR	pressurized water reactor

QA	quality assurance
RH	relative humidity
SCC	stress corrosion cracking
SNF	spent nuclear fuel
SR	site recommendation
SZ	saturated zone
THC	thermal-hydrologic-chemical
TEDE	total effective dose equivalent
TSPA	total system performance assessment
TSPA&I	Total System Performance Assessment and Integration
TSPA-LA	Total System Performance Assessment–License Application
TSPA-SR	Total System Performance Assessment–Site Recommendation
TSPA-VA	Total System Performance Assessment–Viability Assessment
UZ	unsaturated zone
VA	Yucca Mountain Site Characterization Project Viability Assessment
YMP	Yucca Mountain Site Characterization Project

RADIONUCLIDE ABBREVIATIONS

Ac	Actinium
Ar	Argon
Am	Americium
C	Carbon
Ce	Cerium
Cs	Cesium
I	Iodine
Pb	Lead
Ni	Nickel
Np	Neptunium
Pd	Palladium
Pu	Plutonium
Pa	Protactinium
Ra	Radium
Se	Selenium
Sr	Strontium

ACRONYMS AND ABBREVIATIONS (Continued)

Tc	Technetium
Th	Thorium
U	Uranium

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1. INTRODUCTION

This document reports the development of total system performance assessment (TSPA) for the site recommendation (SR). The first section defines the general process involved in developing any TSPA, describes the regulatory requirements for the Total System Performance Assessment-Site Recommendation (TSPA-SR), describes the overall TSPA process as implemented by programs in the United States and elsewhere in the world, and discusses the acceptability of TSPA as a process or tool for analyzing a nuclear waste repository system. It also presents information on previous TSPAs. Section 2 discusses the more specific use of the TSPA process for the TSPA-SR for Yucca Mountain, including approach and methods. Section 3 briefly discusses each of the component models that comprise the TSPA-SR. The TSPA-SR components are: unsaturated zone (UZ) flow, thermal hydrology, in-drift geochemical environment, engineered barrier system environments, waste package and drip shield degradation, waste form degradation, engineered barrier system (EBS) transport, UZ transport, saturated zone (SZ) flow and transport, and biosphere. For each of these components, this section introduces the conceptualization of each individual process, describes the data sources, and discusses model parameter development and computer methods used to simulate each component. Each TSPA component model represents a discrete set of processes. Volcanism is also included in this discussion. Section 4 explains the mechanics of how the individual TSPA components were combined into a nominal case, a disruptive case, and a combined case and provides the probabilistic results for each. In addition, the human intrusion analyses are presented. The section closes with a look at key disruptive events not included in the analyses and an alternative design case. Section 5 addresses sensitivity studies for each of the TSPA components to understand how uncertainty in various parameters within a component affect the TSPA results. Section 5 also contains a description of the probabilistic analyses and results that helps determine the relative importance of the various TSPA components or barriers and the data used to describe the components. Section 6 presents a summary of the findings of the sensitivity studies run on the various components in Section 5, and prioritizes the findings of the entire set of uncertainty and sensitivity studies of the components relative to each other. Section 6 also provides a discussion of factors affecting postclosure performance.

This document procedurally addresses the applicability of *Quality Assurance Requirements and Description* (QARD) (DOE 2000 [149540]) requirements to the work, systems, structures, components, models, analyses, and natural barriers that are discussed in the document. The document was prepared and the development of the model and analyses have been controlled utilizing the current quality assurance (QA) procedures for the project. The QAP-2-0, *Conduct of Activities* evaluation (CRWMS M&O 1999 [119602]), conducted by the Performance Assessment Department, concluded that all activities related to the development of this document or any information contained within it should be conducted utilizing the current QA procedures.

The methods used to control the electronic management of data as required by AP-SV.1Q [153202], *Control of the Electronic Management of Information*, were not specified in the Development Plan. With regard to the development of this report, the control of electronic management of data was evaluated in accordance with YAP-SV.1Q, *Control of the Electronic Management of Data*. The evaluation (CRWMS M&O 2000 [150105]) determined that current work processes and procedures are adequate for the control of electronic management of data for this activity. Though YAP-SV.1Q has been replaced by AP-SV.1Q, this evaluation remains in effect.

This document may be affected by technical product input information that requires confirmation. Any changes to the document that may occur as a result of completing the confirmation activities will be reflected in subsequent revisions. The status of the technical product input information quality may be confirmed by review of the DIRS database.

The document contains numerous appendices to provide additional detail on the TSPA-SR analyses that have been conducted, and to assist the reader in understanding the overall TSPA.

Appendix A–Provides a general glossary as well as a statistical terms glossary for the readers' use.

Appendix B–Provides a description of the feature, event, and process (FEP) database and the approach to development of the database. Tables summarizing the FEPs currently in the database are also provided.

Appendix C–Provides some useful analyses of natural analogs pertinent to the Yucca Mountain repository. In particular, comparisons are provided for radionuclide transport from Peña Blanca and volcanic eruption from Cerro Negro.

Appendix D–Provides useful correlation of the U.S. Nuclear Regulatory Commission's (NRC) Issue Resolution Status Reports (IRSRs) to the TSPA analyses and supporting documentation.

Appendix E–Provides a mapping of the inputs to the TSPA-SR model document (CRWMS M&O 2000 [148384]) from various supporting documents (Analysis Model Reports [AMRs]) in graphical and tabular form.

Appendix F–Provides a synthesis of the major assumptions and conservatisms in the TSPA-SR model. The assumptions may drive the results of the TSPA-SR performance, so it is crucial to understand them. The conservatisms may lead to poorer performance reported from the repository simulations than would be expected for a more realistic portrayal of the particular function involved. Often the conservatisms are utilized due to lack of defensible information concerning a particular process ongoing in the repository system.

Appendix G–Provides the data tracking information for the analyses conducted for the TSPA-SR. This information is listed in terms of data tracking numbers, model simulation run numbers, plot numbers, and so forth to provide ease of traceability of the analyses.

Appendix H–Provides a summary and response to review comments on previous Yucca Mountain TSPA iterations.

1.1 DEFINITION OF PERFORMANCE ASSESSMENT AND TOTAL SYSTEM PERFORMANCE ASSESSMENT

Performance assessment and TSPA are terms with very specific meanings in the high-level radioactive waste (HLW) management community. The process of constructing and implementing a TSPA is often described as a pyramid, where detailed information representing the various processes and components of a total system are distilled and linked into progressively more abstracted models used to analyze system performance.

1.1.1 Explanation of a Total System Performance Assessment

Performance assessment is a method of forecasting how a system, or parts of a system, designed to contain radioactive waste will behave over time. Its goal is to aid in determining whether the system can meet established performance requirements. A TSPA is the subset of performance assessment analyses in which all of the components of a system are linked into a single analysis.

The word forecast, rather than predict, is used to describe the expected outcome of a TSPA. Predict implies inference from facts or accepted laws of nature. Forecast has a similar meaning, but also implies anticipating eventualities and differs from predict in usually being concerned with probabilities instead of certainties. As discussed in Section 1.4, incorporation of probabilities and uncertainty is a critical aspect of TSPA which allows determination of reasonable assurance, as defined by regulatory agencies. However, it must be noted that NRC uses the term predictive models to express what NRC anticipates in proposed 10 CFR Part 63 (64 FR 8640 [101680]). (Note that whenever this document makes direct reference to 10 CFR 63 this document conveys a corresponding reference to Interim Guidance [Dyer 1999 [105655]]) The NRC defines a performance assessment as a probabilistic analysis that:

> (1) identifies the features, events and processes that might affect the performance of the geologic repository; and (2) examines the effects of such features, events, and processes on the performance of the geologic repository; and (3) estimates the expected annual dose to the average member of the critical group as a result of releases from the geologic repository. (10 CFR 63.2)

The process of performance assessment is somewhat different from a safety assessment or a probabilistic risk assessment. Safety assessments use a conservative bounding assessment of the entire system; performance assessments analyze the best understanding of the system and its components (Nuclear Energy Agency (NEA) 1995 [100480], pp. 28 to 36). In a safety assessment, a given process or event is assumed to happen, regardless of the likelihood of its occurrence. A performance assessment incorporates more information than a safety assessment by assuming that some processes or events are more likely to happen than others, and treating them accordingly in mathematical modeling. However, for some processes or events where information is limited, bounding analyses may be used in the performance assessment. The benefit of a performance assessment in this case is that a more realistic and, therefore, more defensible case is used. It must be noted that, in the community of nuclear waste management professionals, the distinction between a safety assessment and a performance assessment has become blurred such that, in informal usage, they are often used interchangeably. However, it is important to differentiate the two philosophies (i.e., use of conservative bounding cases versus use of the most realistic models possible). In addition, a safety case, as made before a licensing authority, could include both safety assessments and performance assessments as defined above.

Probabilistic risk assessment is a term generally applied to safety studies of nuclear power plants or other engineered systems, but it can be applied to any system that could fail in identifiable ways. Although this type of analysis incorporates variations in probability for different processes, the system and the time periods are very different than those used in a performance assessment. A probabilistic risk assessment is usually performed for discrete events of limited duration involving an engineered system and its components. Natural events such as earthquakes and volcanic eruptions are considered initiating events that may have an effect on overall system behavior, but are not a part of the system. The components can be tested on a time scale similar to that for the operational life of the system. Therefore, a set of requirements and specifications for these components is available, and the analyses are performed against criteria that have been, or can be, tested and validated.

A performance assessment treats both the engineered and natural system components. The engineered system is, to some extent, controllable, but the natural system is not. The responses of the total system extend over periods beyond those for which data have been, or can be, obtained.

1.1.2 The Performance Assessment Pyramid

The process for constructing a TSPA is shown in Figure 1.1-1 and described in more detail in Section 1.4. The Performance Assessment Pyramid shows how more detailed underlying information builds the technical basis for the total system models. The breadth of the lowest level of the pyramid represents the complete suite of process and design data and information (i.e., field and laboratory studies that are the first step in understanding the system). The next (higher) level indicates how these data are used to develop conceptual models as well as numerical process models of how the various individual system components are expected to perform under the anticipated repository-relevant conditions. Most of the information at these lower levels (e.g., data, conceptual models, and detailed process models) is synthesized in the Process Model Reports (PMRs) listed below.

- *Biosphere Process Model Report* (CRWMS M&O 2000 [151615])
- *Engineered Barrier System Degradation, Flow, and Transport Process Model Report* (CRWMS M&O 2000 [145796])
- *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332])
- *Integrated Site Model Process Model Report* (CRWMS M&O 2000 [146988])
- *Near Field Environment Process Model Report (PMR)* (CRWMS M&O 2000 [153178])
- *Saturated Zone Flow and Transport Process Model Report* (CRWMS M&O 2000 [145738])
- *Disruptive Events Process Model Report* (CRWMS M&O 2000 [141733])

- *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774])
- *Waste Package Degradation Process Model Report* (CRWMS M&O 2000 [138396]).

The next (higher) level represents the synthesis of information from the lower levels of the pyramid into computer models. At this point, the subsystem behavior may be described by linking models together into representations, as described in Section 3. At this point performance assessment modeling usually begins. The term abstraction is used here to indicate the extraction of essential information. That is, information that is required to enable determination of the effect of a particular process on the overall system performance. In some cases, very little detail is eliminated from the data or the process model to develop the abstraction. Some of the component models are really linkage with a detailed process model. In other cases, the component model may be largely insignificant to performance, and may be reduced to only providing a limited set of information to the TSPA model. The abstraction though, must still represent the characteristics of the component model well enough for the overall TSPA to be a useful representation of the system.

The upper level shows the final level of distillation of information into the most critical aspects necessary to represent the total system. At this point, all of the models are linked together in the TSPA model. These are the models used to forecast total system behavior and estimate the likelihood that the behavior will comply with regulations and ensure long-term safety.

As information flows up the pyramid, it is generally distilled into progressively more simplified forms, or becomes more abstracted, as indicated in Figure 1.1-1. However, abstraction is not synonymous with simplification. If a particular component model can not be simplified without losing essential aspects of the model, it ceases to move up the pyramid and becomes part of the TSPA calculation tool. Thus, an abstracted model in a TSPA may take the form of something as simple as a table of values that were calculated using a complex computer model. The abstraction may also take the form of a fully three-dimensional computer simulation. It must be noted that even the most complex models of specific processes are still an abstraction of reality.

There are also some considerations that dictate the level of complexity used to represent a process. One is the sensitivity of the results of the TSPA to that particular process. The more sensitive the process or parameter, the more detailed the model representation tends to be. However, the degree of complexity is also limited by the state of knowledge concerning the model. It is very important not to misrepresent the degree of understanding about a process by embedding it in an overly complicated computer model.

Another aspect of the development of the TSPA model is the use of conservatism in the assumptions chosen to assist in development of the model. These conservative assumptions are utilized for several reasons including lack of data, incomplete knowledge of the uncertainty of a feature, and inability either from resources or timing to defend potentially positive performance, where "defensibility" in the licensing arena is a project objective. The major conservatism in the assumptions is presented in Appendix F. Conservative is used here to indicate that the assumption or model used may underestimate the positive performance of a particular part of the repository system, but allows a more defensible position for the analyses. Alternatively, conservatism may cause negative performance to be overstated which is potentially problematic if additional resources are required to bolster engineered systems to overcompensate for the conservatism embedded in the analyses. The project is undergoing a significant review of conservatism, and may cause some redirection in this area for future iterations of TSPA.

Subsystem level conservatism may not significantly impact the TSPA-SR model performance. However, there is a risk of compounding conservatism if multiple aspects of the system are given conservative or under performing characteristics that may lead to significantly degraded overall performance. This can be evaluated using less conservative assumptions for features of the repository system that are known to be important to performance.

1.2 OBJECTIVES OF TOTAL SYSTEM PERFORMANCE ASSESSMENT FOR THE SITE RECOMMENDATION

The objective of the TSPA for the SR, based on design concepts and scientific data and analysis currently available, is to provide an assessment of repository performance at the potential Yucca Mountain Site as part of the site recommendation process. The scope of the TSPA is guided by technical requirements proposed by the NRC at proposed 10 CFR Part 63 (64 FR 8640 [101680]), and radiation protection standards proposed by the EPA at proposed 40 CFR 197 (64 FR 46976 [105065]). The analyses must be traceable and transparent.

Assessing the performance of the system requires the following:

- Assimilating all the available scientific data and analyses that describe the geological setting into which the design concept is to be placed
- Defining the design concept that is to be used
- Describing the behavior of the potential repository system in a traceable, transparent manner
- Identifying the performance standards by which the TSPA will be judged.

The total system is comprised of geological and engineering components. Therefore, the TSPA uses the available scientific information about naturally occurring physical and chemical processes at the Yucca Mountain site. In addition, the TSPA includes the design concepts and scientific information about physical and chemical processes involving the engineered components.

The current overall system-performance standards utilized for this analysis of the potential Yucca Mountain repository are found in proposed 10 CFR Part 63 (64 FR 8640 [101680]) and proposed 40 CFR Part 197 (64 FR 46976 [105065]).

The U.S. Nuclear Regulatory Commission (NRC), in proposed 10 CFR Part 63 (64 FR 8640 [101680]), provided a measure of system performance that limits the annual committed dose from radionuclides released from the facility to the average member of the critical group residing in a farming community located 20 km downgradient from the potential repository. This distance was chosen to correspond to Lathrop Wells, the closest existing public or private well to the site, near the intersection of U.S. Highway 95 and Nevada State Route 373. The controlled area boundary for the DOE Nevada Test Site (NTS) also is approximately 20 km from the potential repository. Per proposed 10 CFR Part 63 (64 FR 8640 [101680]), the analyses must demonstrate "reasonable assurance" that the expected dose to the average member of the critical group will not exceed 25 mrem in 10,000 years. Per proposed 40 CFR Part 197 (64 FR 46976 [105065]), the DOE must demonstrate a "reasonable expectation" that a dose of 15 mrem/yr will not be attained in 10,000 years.

While the regulations require evaluation of a 10,000-year time period, the TSPA analyses will evaluate the consequences caused by the potential repository beyond that period. The analyses are extended to 100,000 and 1 million years in determining when the peak radionuclide doses or peak risk occurs. The analyses beyond 10,000 years are providing information to support assessments contained in the Program's Environmental Impact Statement (EIS) that will accompany the Site Recommendation (SR) to the President.

Although the goal of the TSPA is to provide a quantitative assessment of the performance of the potential repository system, it is important to recognize the uncertainties inherent in such analyses. The U.S. Environmental Protection Agency (EPA) and the NRC have recognized the care required in defining the degree of confidence needed from the analyses. EPA stated that (proposed 40 CFR 197.14(a) [64 FR 46976 [105065]]):

> Reasonable expectation: (a) requires less than absolute proof because absolute proof is impossible to attain for disposal due to the uncertainty of projecting long term performance; (b) is less stringent than the reasonable assurance concept that NRC uses to license nuclear power plants; (c) takes into account the inherently greater uncertainties in making long-term projections of the performance of the Yucca Mountain disposal system; (d) does not exclude important parameters from assessments and analyses simply because they are difficult to precisely quantify to a high degree of confidence; and (e) focuses performance assessments and analyses upon the full range of defensible and reasonable parameter distributions rather than only upon extreme physical situations and parameter values.

NRC also underscored this point in its discussion of reasonable assurance (10 CFR Part 60 [103540]):

> The Commission anticipates that licensing decisions will be complicated by the uncertainties that are associated with predicting the behavior of a geologic repository over the thousands of years during which HLW may present hazards to public health and safety.

These inherent uncertainties were recognized in developing the analysis tools that are described in Sections 2.2 and Section 3 of this volume. The potential effects of many of these uncertainties are presented in Section 5.

Given the uncertainty involved in a postclosure performance assessment, an important goal is to produce a transparent document describing the assumptions, the intermediate steps, the results, and the conclusions of the analyses. The U.S. Nuclear Waste Technical Review Board (NWTRB) states that "transparency is the ease of understanding the process by which a study was carried out, which assumptions are driving the results, how they were arrived at, and the rigor of the analyses leading to the results" (NWTRB 1998 [100482], p. 21). The TSPA Peer Review Panel notes that "transparency is achieved when a reader or reviewer has a clear picture of what was done in the analysis, what the outcome was, and why" (Budnitz et al. 1997 [100427], pp. 9 to 10).

For the reader to have confidence in the analyses, the presentation must illustrate with sufficient clarity the following attributes:

- The conceptual basis for the individual components in the quantitative analyses (i.e., how the system is intended to work, which is presented in Section 2.1)
- How individual components are combined into an assessment of system behavior (Sections 2.2 and 4.1)
- The scientific understanding used to develop the quantitative analysis tools that describe the system's expected evolution (Sections 3.1 to 3.10)
- The system's expected evolution as defined by the spatial and temporal response of the system to waste emplacement (Sections 4.1 and 4.2)
- Uncertainty in the system's expected evolution and the significance of that uncertainty to the system-performance goals (Section 5).

1.3 REGULATORY REQUIREMENTS FOR THE TOTAL SYSTEM PERFORMANCE ASSESSMENT FOR THE SITE RECOMMENDATION

The regulatory requirements for the TSPA-SR have been almost three decades in the making. In 1978, an Interagency Review Group on Nuclear Waste Management began coordinating the interrelated activities already underway within the EPA, the NRC, and the DOE. Congress passed the Nuclear Waste Policy Act of 1982 [100014], Public Law No. 97-425, to establish the national policy. The three rules pertinent to this TSPA-SR were proposed in rulemaking proceedings for public review and comment in 1999. These rules, proposed 40 CFR Part 197 (64 FR 46976 [105065]), proposed 10 CFR Part 63 (64 FR 8640 [101680]), and proposed 10 CFR Part 963 (64 FR 67054 [124754]), define the relationships among the EPA, NRC, and DOE regarding the potential Yucca Mountain repository. This section of the TSPA-SR summarizes the developments leading to these regulations, and focuses on the resulting requirements for TSPA. Figure 1.3-1 illustrates the timeline of pertinent lawmaking and rulemaking events. The section discusses the method established by the NRC to track important issues in potential repository performance, with emphasis on the TSPA. A regulatory framework is provided for the analyses and results presented in this TSPA-SR.

1.3.1 Nuclear Waste Policy Act of 1982, as Amended

The foundation for the Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., was laid in the Nuclear Waste Policy Act of 1982 [100014], Public Law No. 97-425, which selected permanent disposal in deep geologic repositories to "provide a reasonable assurance that the public and the environment will be adequately protected from the hazards posed by [SNF and HLW]" (Nuclear Waste Policy Act of 1982 [100014], Public Law No. 97-425). The act established the DOE authority and responsibility for siting, constructing, and operating such repositories. It also assigned regulatory roles to the EPA and the NRC. In the Nuclear Waste Policy Amendments Act of 1987 [100016], Public Law No. 100-203, Congress amended the NWPA to designate Yucca Mountain as the only site to be characterized. Congress again amended the NWPA (DOE 1995 [122137]) in the Energy Policy Act of 1992 [100017], Public Law No. 102-486, That act also directed the EPA to promulgate radiation protection standards specifically for Yucca Mountain and directing the NRC to modify its technical requirements and criteria (10 CFR Part 60 [48 FR 28194 [100475]]) to be consistent with the new EPA standards. Congress required the EPA to contract with the National Academy of Sciences (NAS) to study radiation protection standards before setting the new standard. The new EPA standards are required to be based upon and consistent with the NAS findings and recommendations.

In *Technical Bases for Yucca Mountain Standards* (National Research Council 1995 [100018]), the NAS recommended an approach and content significantly different from those previously adopted by the EPA and the NRC. The NAS determined that analyses of potential repository behavior covering thousands of years are scientifically justifiable and possible. They found that a health standard based on risk to individuals of adverse health effects from releases from the potential repository (instead of the generic standards at 40 CFR Part 191 (58 FR 66398 [107802]), which contain both individual dose and release limits) would adequately protect the general public. They also found that predictions regarding the probability that a potential repository will be breached by human intrusion during a period of 10,000 years cannot be scientifically supported. The NAS recommended that the EPA include in its regulations a stylized human intrusion event to provide insight into the degree to which an intrusion would degrade the performance of a potential repository. The NAS concluded that the performance of the total system, rather than that of its individual elements in isolation, is crucial in the context of a risk-based standard, because subsystem performance requirements could result in a deficient potential repository design even if each subsystem element meets or exceeds the performance standard. The TSPA approach has been employed at Yucca Mountain since 1991.

1.3.1.1 U.S. Environmental Protection Agency Authority and Responsibility

The Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., as originally enacted directed the EPA to promulgate generic radiation standards, thus ensuring that the regulatory requirements for a potential repository would be set independently of potential repository development. In the Energy Policy Act of 1992 [100017], Public Law No. 102-486, Congress separated the EPA's health-based standard for Yucca Mountain from the generic EPA standards promulgated at 40 CFR Part 191 (58 FR 66398 [107802]). Congress also gave the EPA sole authority to set public health and safety radiation standards for Yucca Mountain. The EPA published proposed standards at proposed 40 CFR Part 197 (64 FR 46976 [105065]) on August 27, 1999.

1.3.1.2 U.S. Nuclear Regulatory Commission Authority and Responsibility

Under the Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., as amended, the NRC was directed to establish technical requirements and criteria, consistent with any comparable EPA standards, providing for the use of a system of multiple barriers and, if deemed appropriate, restricting retrievability. The Energy Policy Act of 1992 [100017], Public Law No. 102-486, requires the NRC to conduct a licensing proceeding before authorizing the construction of a potential repository. Separate licensing proceedings will also be required for authorization of operation and closure of the potential repository. The Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., as amended also required the NRC to modify its technical requirements and criteria within one year after the establishment of final EPA standards. The proposed regulations, 10 CFR Part 63 (64 FR 8640 [101680]) were published February 22, 1999. The NRC's technical requirements and criteria for construction, operation, and closure of a potential geologic repository will have a broader role for Yucca Mountain than just to implement the EPA standards. Those regulations will govern the licensing process if the Yucca Mountain site is recommended by the Secretary to the President, approved by the President, and is designated by Congress under the Nuclear Waste Policy Act of 1982, 42 U.S.C. 10101 et seq [101681]. However, the EPA standards drive the NRC performance objectives that determine the complexity of this TSPA-SR.

1.3.1.3 U.S. Department of Energy Authority and Responsibility

After Congress assigned to the DOE the responsibility to dispose of SNF and HLW in geologic repositories, the DOE promulgated guidelines for recommending candidate sites for site characterization at proposed 10 CFR Part 960 (49 FR 47714 [100562]). The *Site Characterization Plan Overview, Yucca Mountain Site, Nevada Research and Development Area, Nevada* (DOE 1988 [100281]) was required to include criteria for determining the suitability of a site for the location of a potential repository. In the Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., Congress also addressed site recommendation, approval, and construction authorization, which can only proceed as site characterization activities near completion. After Congress approves the site, the DOE must submit an application to the NRC for a construction authorization. The DOE has proposed new guidelines at proposed 10 CFR Part 963 (64 FR 67054 [124754]), published November 30, 1999. These guidelines provide detailed requirements for the TSPA, are consistent with the proposed NRC regulations, and adhere to the applicable radiation protection standards.

1.3.1.4 Synopsis of Performance Measures for the Postclosure Period

Whether the site can be determined to be suitable for recommendation depends on the estimated capability of the potential repository to satisfy the radiation protection standards. The NRC rule was developed in parallel with the EPA standards. As a result, the EPA and the NRC proposed different dose limits, the EPA proposed a ground water protection standard, and different approaches were taken for consideration of human intrusion. The DOE has, therefore, proposed to base its suitability determination on the "applicable radiation protection standard," i.e., the final EPA standard as implemented by the NRC (64 FR 67054 [124754], pp. 67074 and 67075). Tables 1.3-1, 1.3-2, and 1.3-3 present the proposed performance measures for the postclosure period contained in the three proposed rules.

Table 1.3-1. Proposed Performance Measures for Postclosure–Undisturbed Performance

Rule	Performance Measure
EPA Proposed 40 CFR PART 197	Individual-Protection Standard. 197.20 The DOE must demonstrate, using performance assessment, that there is a reasonable expectation that for 10,000 years following disposal the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microsieverts (15 mrem) from releases from the undisturbed Yucca Mountain disposal system. The DOE's analysis must include all potential pathways of radionuclide transport and exposure. Ground Water Protection Standards. 197.35. In its license application to NRC, DOE must provide a reasonable expectation that, for 10,000 years of undisturbed performance after disposal, releases of radionuclides from radioactive material in the Yucca Mountain disposal system will not cause the level of radioactivity in the representative volume of ground water at the point of compliance to exceed Combined 226Ra and 226Ra: 5 pCi/L including natural background Gross alpha activity (including 226Ra but excluding radon and uranium): 15 pCi/L including natural background Combined beta and photon emitting radionuclides: 40 mSv/yr. (4 mrem/yr.) to the whole body or any organ.
NRC Proposed 10 CFR PART 63	63.113 Performance Objective For The Geologic Repository After Permanent Closure. (a) The geologic repository shall include multiple barriers, consisting of both natural barriers and an engineered barrier system. (b) The engineered barrier system shall be designed so that, working in combination with natural barriers, the expected annual dose to the average member of the critical group shall not exceed 0.25 mSv (25 mrem) TEDE (total effective dose equivalent) at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository. (c) The ability of the geologic repository to limit radiological exposures to those specified in 63.113(b) shall be demonstrated through a performance assessment that meets the requirements specified at 63.114, uses the reference biosphere and critical group specified at 63.115, and excludes the effects of human intrusion.
DOE Proposed 10 CFR PART 963	963.15 Postclosure Suitability Determination. DOE will apply the method and criteria described in Sections 963.16 and 963.17 to evaluate the suitability of the Yucca Mountain site for the postclosure period. If DOE finds that the results of the total system performance assessments conducted under 963.16(a)(1) show that the Yucca Mountain site is likely to meet the applicable radiation protection standard, DOE may determine the site suitable for the postclosure period. 963.16 Postclosure Suitability Evaluation Method. (a) DOE will evaluate postclosure suitability using the [TSPA] method.... (1) DOE will conduct a [TSPA] to evaluate the ability of the geologic repository to limit radiological exposures in the case where there is no human intrusion into the repository. DOE will model the performance of the geologic repository at the Yucca Mountain site using the method described in 963.16(b) and the criteria in Sec 963.17, excluding the criterion in 963.17(b)(4). DOE will consider the performance of the system in terms of the criteria to evaluate whether the geologic repository is likely to comply with the applicable radiation protection standard.

Sources: Proposed 10 CFR Part 63 (64 FR 8640 [101680]); proposed 40 CFR Part 197 (64 FR 46976 [105065]); proposed 10 CFR Part 963 (64 FR 67054 [124754])

NOTES: [1] Undisturbed performance means that human intrusion or the occurrence of "unlikely," disruptive, natural processes and events do not disturb the disposal system (64 FR 46976 [105065], p. 47014). The DOE defined disruptive features, events, and processes (DFEPs) to mean FEPs having a probability of occurrence during the period of performance of less than 1.0 but greater than 10^{-4} in 10^4 years (CRWMS M&O 1999 [123126], App. A).

[2] The EPA proposes to interpret the term "undisturbed," used by the NAS in its recommendations, to mean that the Yucca Mountain disposal system would not be disturbed by human intrusion, but could be disturbed by other processes or events that are "likely" to occur (64 FR 46976 [105065], p. 46998).

Table 1.3-2. Proposed Performance Measures for Postclosure Period–Disturbed Performance

Rule	Performance Measure
EPA Proposed 40 CFR PART 197	Individual-Protection Standard. 197.20 The DOE must demonstrate, using performance assessment, that there is a reasonable expectation that for 10,000 years following disposal the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microsieverts (15 mrem) from releases from the undisturbed Yucca Mountain disposal system. The DOE's analysis must include all potential pathways of radionuclide transport and exposure.
NRC Proposed 10 CFR PART 63	63.113 Performance Objective For The Geologic Repository After Permanent Closure. (a) The geologic repository shall include multiple barriers, consisting of both natural barriers and an engineered barrier system. (b) The engineered barrier system shall be designed so that, working in combination with natural barriers, the expected annual dose to the average member of the critical group shall not exceed 0.25 mSv (25 mrem) TEDE [total effective dose equivalent] at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository. (c) The ability of the geologic repository to limit radiological exposures to those specified in 63.113(b) shall be demonstrated through a performance assessment that meets the requirements specified at 63.114, uses the reference biosphere and critical group specified at 63.115, and excludes the effects of human intrusion.
DOE Proposed 10 CFR PART 963	963.15 Postclosure Suitability Determination. DOE will apply the method and criteria described in Sections. 963.16 and 963.17 to evaluate the suitability of the Yucca Mountain site for the postclosure period. If DOE finds that the results of the total system performance assessments conducted under [§963.16(a)(1)] show that the Yucca Mountain site is likely to meet the applicable radiation protection standard, DOE may determine the site suitable for the postclosure period. 963.16 Postclosure Suitability Evaluation Method. (a) DOE will evaluate postclosure suitability using the [TSPA] method.... (1) DOE will conduct a [TSPA] to evaluate the ability of the geologic repository to limit radiological exposures in the case where there is no human intrusion into the repository. DOE will model the performance of the geologic repository at the Yucca Mountain site using the method described in 963.16(b) and the criteria in Sec 963.17, excluding the criterion in 963.17(b)(4). DOE will consider the performance of the system in terms of the criteria to evaluate whether the geologic repository is likely to comply with the applicable radiation protection standard. 963.17(b) Postclosure suitability criteria. DOE will evaluate the postclosure suitability of a geologic repository at the Yucca Mountain site using criteria that consider disruptive processes and events important to the total system performance of the geologic repository. The applicable criteria related to disruptive processes and events include: (1) Volcanism–or example, the probability and potential consequences of a volcanic eruption intersecting the repository; (2) Seismic events–for example, the probability and potential consequences of an earthquake on the underground facilities or hydrologic system; (3) Nuclear criticality–for example, the probability and potential consequences of a self-sustaining nuclear reaction as a result of chemical or physical processes affecting the waste either in or after release from breached waste packages.

Sources: Proposed 10 CFR Part 63 (64 FR 8640 [101680]); proposed 40 CFR Part 197 (64 FR 46976 [105065]); proposed 10 CFR Part 963 (64 FR 67054 [124754])

Table 1.3-3. Proposed Performance Measures for the Postclosure Period–Human Intrusion Case

Rule	Performance Measure
EPA Proposed 40 CFR PART 197	Human Intrusion Standard. Alternative 1 for 197.25: The DOE must demonstrate that there is a reasonable expectation that for 10,000 years following disposal the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microsieverts (15 mrem) as a result of a human intrusion. The DOE's analysis of human intrusion must include all potential environmental pathways of radionuclide transport and exposure. Alternative 2 for 197.25: The DOE must determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion (see 197.26) could occur without recognition by the drillers. The DOE must: (a) Demonstrate that there is a reasonable expectation that the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microsieverts (15 mrem) as a result of a human intrusion, if complete waste package penetration can occur at or before 10,000 years after disposal. The analysis must include all potential environmental pathways of radionuclide transport and exposure; and (b) Include the results of the analysis and its bases in the environmental impact statement for Yucca Mountain as an indicator of long-term disposal system performance, if the intrusion cannot occur before 10,000 years after disposal.
NRC Proposed 10 CFR PART 63	63.113 Performance objective for the geologic repository after permanent closure. (d) The ability of the geologic repository to limit radiological exposures to those specified in §63.113(b), in the event of limited human intrusion into the engineered barrier system, shall be demonstrated through a separate performance assessment that meets the requirements specified at 63.114 and uses the reference biosphere and critical group specified at 63.115. For the assessment required by this paragraph, it shall be assumed that the human intrusion occurs 100 years after permanent closure and takes the form of a drilling event that results in a single, nearly vertical borehole that penetrates a waste package, extends to the saturated zone, and is not adequately sealed.
DOE Proposed 10 CFR PART 963	963.15 Postclosure suitability determination. DOE will apply the method and criteria described in Sections 963.16 and 963.17 to evaluate the suitability of the Yucca Mountain site for the postclosure period. If DOE finds that the results of the [TSPAs] conducted under [963.16(a)(2)] show that the Yucca Mountain site is likely to meet the applicable radiation protection standard, DOE may determine the site suitable for the postclosure period. 963.16 Postclosure Suitability Evaluation Method. (a)(2) Consistent with applicable NRC regulations regarding a stylized human intrusion case, DOE will conduct a [TSPA] to evaluate the ability of the geologic repository to limit radiological exposures in a stylized limited human intrusion case. DOE will model the performance of the geologic repository at the Yucca Mountain site using the method described in 963.16(b) and the criteria in Sec 963.17. DOE will consider the performance of the system in terms of the criteria to evaluate whether the geologic repository is likely to comply with the applicable radiation protection standard. The human intrusion evaluation under this paragraph will be separate from the evaluation conducted under 963.16(a)(1). 963.17 Postclosure Suitability Criteria. (b) DOE will evaluate the postclosure suitability of a geologic repository at the Yucca Mountain site using criteria that consider disruptive processes and events important to the total system performance of the geologic repository. The applicable criteria related to disruptive processes and events include: (b)(4) Inadvertent human intrusion–for example, consequences to repository system performance following a stylized human intrusion scenario.

Sources: Proposed 10 CFR Part 63 (64 FR 8640 [101680]); proposed 40 CFR Part 197 (64 FR 46976 [105065]); proposed 10 CFR Part 963 (64 FR 67054 [124754])

These performance measures define how robust the combined engineered and natural barrier systems must be to protect the public. The regulatory language also prescribes how the TSPA must analyze the potential repository's performance to demonstrate that robustness. TSPA is an inherently complex, multidisciplinary analysis that evaluates movement of radionuclides from the disposal system into the environment. Hence, cognizance of the requirements and criteria for demonstrating performance is necessary to frame the assessment. Cognizance of the regulatory objectives is necessary to evaluate the adequacy of the TSPA for supporting a decision on site recommendation. This TSPA-SR is concerned with the requirements and criteria for analyzing the performance of the potential repository, and with the radiation protection standards to the extent that they dictate where and how that performance will be analyzed.

1.3.2 Proposed 40 CFR Part 197: Environmental Radiation Protection Standards for Yucca Mountain, Nevada

The DOE is the only entity directly regulated by proposed 40 CFR Part 197 (64 FR 46976 [105065]); the NRC is affected because the Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., requires NRC's licensing regulation to be consistent with the EPA's final standards. Although separated from the generic standards, the proposed 40 CFR Part 197 (64 FR 46976 [105065]) reflects the experiences of the EPA in setting and implementing radiation protection standards for a potential geologic repository (e.g., Figure 1.3-1 shows the EPA rulemaking activities for the Waste Isolation Pilot Plant). This discussion covers the postclosure requirements.

In 1985, the EPA established generic standards for the management, storage, and disposal of SNF, HLW, and transuranic radioactive waste at 40 CFR Part 191 (50 FR 38066 [100495]). In 1987, the U.S. Court of Appeals vacated the disposal standards and remanded them to the EPA (*Natural Resources Defense Council, Inc. v. U.S. Environmental Protection Agency* 1987 [149706]). In 1992, the Waste Isolation Pilot Plant Land Withdrawal Act [131959], Public Law No. 102-579, reinstated the 40 CFR Part 191 (58 FR 66398 [107802]) disposal standards, requiring the EPA to replace those that were the specific subject of the remand. That act also exempted the Yucca Mountain site from the 40 CFR Part 191 (63 FR 27354 [151707]) disposal standards and designated the EPA as regulator for the Waste Isolation Pilot Plant. The EPA issued the final disposal standards at 40 CFR Part 191 in 1993 (58 FR 66398 [107802]). The court's concerns were addressed by conforming the groundwater protection requirements to the Safe Drinking Water Act [103937], 42 U.S.C. 300f et seq. Criteria for the certification of the Waste Isolation Pilot Plant were promulgated at 40 CFR Part 191 (61 FR 5224 [107682]) in 1996, and the EPA certified the Waste Isolation Pilot Plant in 1998 (by amending 40 CFR Part 191 [63 FR 27354 [151707]]). In contrast, the NRC will promulgate and implement procedures and requirements for the licensing of the potential Yucca Mountain repository, including requirements for compliance with the EPA standards.

Subpart B of proposed 40 CFR Part 197 (64 FR 46976 [105065]) contains the environmental standards for the disposal of radioactive waste in Yucca Mountain by the DOE. The NRC will determine compliance with Subpart B based upon the results of the DOE's performance assessments projecting the performance of the potential Yucca Mountain repository for 10,000 years after disposal. The DOE must demonstrate to the NRC that there is a reasonable expectation of compliance with Subpart B before the NRC can issue a license. The performance measures for Subpart B are contained in the Individual-Protection Standard (§197.20), the Human Intrusion Standard (§197.25), and the Ground Water Protection Standards (§197.35). These three standards prescribe the analyses that must be included in the TSPA.

For individual protection, the DOE must demonstrate, using performance assessment, a reasonable expectation that, for 10,000 years following disposal, the reasonably maximally exposed individual (proposed 40 CFR 197.21 [64 FR 46976 [105065]]) is safe (Tables 1.3-1 and 1.3-2). The analysis must include all potential pathways of radionuclide transport and exposure.

The consideration of human intrusion into the potential repository, as defined in proposed 40 CFR 197.26 (64 FR 46976 [105065]), must also demonstrate a reasonable expectation that following disposal the reasonably maximally exposed individual is safe (Table 1.3-3). The analysis must include all potential environmental pathways of radionuclide transport and exposure. Two alternatives for considering the time element are proposed. In the first, the time element is simply 10,000 years. In the second alternative, the DOE must determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion by drilling could occur without recognition by the drillers.

Protecting ground water requires that the DOE must provide to the NRC a reasonable expectation that, for 10,000 years of undisturbed performance after disposal, releases of radionuclides from the disposal system will not cause the level of radioactivity in the representative volume (proposed 40 CFR 197.36 [64 FR 46976 [105065]]) of ground water at the point of compliance to exceed the limits specified (see Table 1.3-1).

The EPA defines *reasonable expectation* (proposed 40 CFR 197.14 [64 FR 46976 [105065]]) to mean that the NRC "is satisfied that compliance will be achieved based upon the full record before it." The EPA further specifies that reasonable expectation requires less than absolute proof because absolute proof is impossible to attain for disposal due to the uncertainty of projecting long-term performance. Reasonable expectation is seen by the EPA as "less stringent than the *reasonable assurance* concept that the NRC uses to license nuclear power plants" (see Section 1.3.3.2). The EPA intends reasonable expectation to take into account the inherently greater uncertainties in making long-term projections of the performance of the Yucca Mountain disposal system. However, important parameters should not be excluded from assessments and analyses simply because they are difficult to quantify precisely to a high degree of confidence. Moreover, the EPA intends for reasonable expectation to focus performance assessments and analyses upon the full range of defensible and reasonable parameter distributions, rather than only upon extreme physical situations and parameter values (64 FR 46976 [105065]).

Human society, biology, and knowledge will be assumed unchanging during the 10,000 years after the license submission to the NRC. However, factors related to the geology, hydrology, and climate of the site must be varied based on environmentally protective but reasonable scientific predictions of the changes that could affect the Yucca Mountain disposal system over the next 10,000 years (64 FR 46976 [105065]).

The DOE must also calculate the peak dose to the reasonably maximally exposed individual that would occur after 10,000 years following disposal, but within the period of geologic stability. While no regulatory standard applies to the results of this analysis, the DOE must include the results and their bases in the Environmental Impact Statement (EIS) for Yucca Mountain as an indicator of long-term disposal system performance (64 FR 46976 [105065]).

Performance must be assessed at the point of compliance. The EPA proposed four alternative definitions, numbers one and four of which place the point of compliance at any point on the boundary of the controlled area. Two alternative definitions of *controlled area* (proposed 40 CFR 197.12 [64 FR 46976 [105065], p. 47013]) are proposed. Alternative 1 limits the controlled area to no more than 100 square kilometers. Alternative 2 allows the DOE to include in the controlled area any contiguous area within the boundary of the NTS. One of these two alternatives will be applied if the EPA sets the "point of compliance" (proposed 40 CFR 197.37 [64 FR 46976 [105065], p. 47016]) at any point on the boundary of the controlled area, otherwise the concept of controlled area will not appear in the final rule) (proposed 40 CFR 197.12 [64 FR 46976 [105065], p. 47013]). Alternative 2 places the point of compliance at any point within a half-kilometer radius of the intersection of U.S. Route 95 and Nevada State Route 373. Alternative 3 places the point of compliance within the town of Amargosa Valley, Nevada, more specifically within the area bounded by Frontier Street on the north, Nevada State Route 373 on the east, the Nevada-California border on the south-southwest, and Casada Way on the west. However, if the NRC identifies another location about 20 kilometers (alternative three) from the center of the potential repository footprint where the representative volume would have a higher concentration of radionuclides that were released from the repository, the NRC must specify that location as the point of compliance (64 FR 46976 [105065]).

Performance assessments need not consider processes or events estimated to have less than one chance in 10,000 of occurring within 10,000 years of disposal. The EPA proposes to allow the NRC to change this limit to exclude slightly higher probability events. If so, the performance assessments need not evaluate, in detail, the impacts resulting from any processes and events or sequences of processes and events with a higher chance of occurrence if the results of the performance assessments would not be changed significantly (64 FR 46976 [105065]).

1.3.3 Proposed 10 CFR Part 63: Disposal of High-Level Radioactive Wastes in a Potential Geological Repository at Yucca Mountain, Nevada

The DOE is the only entity directly regulated by the proposed 10 CFR Part 63 (64 FR 8640 [101680], p. 8659). Note that whenever this document makes direct reference to proposed 10 CFR 63, this document conveys a corresponding reference to DOE's Interim Guidance (Dyer 1999 [105655]). This rule specifies how the NRC will carry out its licensing obligations under the Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq., and how it will implement proposed 40 CFR Part 197 (64 FR 46976 [105065]) for the potential repository. The NRC criteria address the performance of the potential repository system, which must comprise both natural and engineered barriers. Also included are licensing procedures, criteria for public participation, records and reporting, monitoring and testing programs, performance confirmation, QA, personnel training and certification, and emergency planning. The proposed criteria will apply specifically and exclusively to Yucca Mountain. The proposed NRC rulemaking also affects 10 CFR Parts 2 [100502], 19 [103585], 20 [104787], 21 [140852], 30 [150331], 40 [151723], 51 [144582], 60 [103540], and 61 (64 FR 8640 [105065], p. 8658). Parts 2 [100502], 19 [103585], 20 [104787], 21 [140852], and 51 [144582] would be amended to apply to Part 63. Parts 30, 40, and 61 would be amended to exempt DOE for activities related to Part 63. Part 60 would be amended to clarify that it does not apply to, and cannot be the subject of litigation in, any NRC licensing proceeding for a potential repository at Yucca Mountain (64 FR 8640 [101680], p. 8640).

Generic regulations at 10 CFR Part 60 [103540] govern the licensing of the DOE to receive and possess source, special nuclear, and byproduct material at a geologic repository that is sited, constructed, and operated under the Nuclear Waste Policy Act of 1982 [101681], as amended, 42 U.S.C. 10101 et seq. Figure 1.3-1 includes an illustration of the evolution of this rule. The NRC's technical criteria assumed that the EPA standards would limit only cumulative radionuclide releases from a geologic repository. In 1985, the EPA issued final standards in 40 CFR Part 191 (50 FR 38066 [100495]) containing cumulative release limits, but also containing criteria for individual and groundwater protection. Although the NRC proposed "conforming amendments" to incorporate the EPA standards into the NRC regulations (51 FR 22288 [151059]; 64 FR 8640 [101680], p. 8640), they were abandoned in 1987 when the EPA standards were vacated by the U.S. Court of Appeals (64 FR 8640 [101680], p. 8640).

During the years since the initial technical criteria were promulgated, the technical methods for performance assessment have evolved significantly. "The implementation of these new methods for Yucca Mountain will avoid the imposition of unnecessary, ambiguous, or potentially conflicting criteria that could result from the application of proposed 10 CFR Part 60" (64 FR 8640 [101680], p. 8641). This discussion focuses on those parts of proposed 10 CFR Part 63 (64 FR 8640 [101680], p. 8641) that differ from 10 CFR Part 60 (46 FR 13971 [151057], 48 FR 28194 [100475], 50 FR 29641 [151083], 51 FR 27158 [151058], 54 FR 27864 [151082], 61 FR 64257 [104190]) and that apply to the TSPA analysis of postclosure performance.

1.3.3.1 Proposed Part 63 Subpart B–Licenses

Site characterization must be conducted prior to submittal of an application and in a manner that limits adverse effects on the performance of the potential geologic repository. The DOE must submit semiannual reports on the progress of site characterization. NRC staff may visit, inspect, and observe site characterization activities and comment on any aspect of site characterization and performance assessment. The License Application (LA) must include general information and a safety analysis report, and be accompanied by an EIS. Subpart B describes the information to be included in the safety analysis report (64 FR 8640 [101680]). The performance assessment, an assessment of how the FEPs of the site affect waste isolation, and an assessment of the responses of the natural systems to thermal loading are major portions of the safety analysis report. These analyses are integral to this TSPA-SR.

1.3.3.2 Proposed Part 63 Subpart E–Technical Criteria

Subpart E contains proposed performance objectives through permanent closure (preclosure) and after permanent closure (postclosure). It contains the requirements for the analyses used to demonstrate compliance with the performance objectives. Subpart E requires that compliance be demonstrated in the context of safety analyses of total system performance (64 FR 8640 [101680]).

The NRC recognized (proposed 10 CFR 63.101[a][2] [64 FR 8640 [101680]]) that complete assurance that the requirement will be met is not achievable. The general standard that the NRC requires is a *reasonable assurance*, based on the record before it, that the performance objective proposed in 10 CFR 63.113 (64 FR 8640 [101680]) will be met (see Section 1.3.2).

Proof that the potential geologic repository will be in conformance with the objective for postclosure performance is not to be had in the ordinary sense of the word because of the uncertainties inherent in the understanding of the evolution of the geologic setting, biosphere, and EBS. For such long-term performance, what is required is reasonable assurance, making allowance for the time period, hazards, and uncertainties involved, that the outcome will be in conformance with the objective for postclosure performance of the potential geologic repository. Demonstrating compliance will involve the use of complex predictive models that are supported by limited data from field and laboratory tests, site-specific monitoring, and natural analog studies that may be supplemented with prevalent expert judgment. Further, in reaching a determination of reasonable assurance, the Commission may supplement numerical analyses with qualitative judgments including, for example, consideration of the degree of diversity among the multiple barriers as a measure of the resiliency of the potential geologic repository (64 FR 8640 [101680], p. 8674).

The performance objective for the potential geologic repository after permanent closure, proposed 10 CFR 63.113 (64 FR 8640 [101680], p. 8676), requires the DOE to include a system of multiple barriers, comply with the individual annual dose limit, conduct a performance assessment, and assess the consequences of a specified human intrusion event. Requirements for the performance assessment to demonstrate compliance with the individual dose limit are shown in Table 1.3-4. Characteristics of the reference biosphere and critical group for the performance assessment are shown in Table 1.3-5. These requirements and characteristics define the scope of the TSPA-SR.

Table 1.3-4. Proposed U.S. Nuclear Regulatory Commission Requirements for Performance Assessment

Section of Proposed 10 CFR Part 63	Requirements
63.114(a)	Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site, and the surrounding region to the extent necessary, and information on the design of the engineered barrier system, used to define parameters and conceptual models used in the assessment.
63.114(b)	Account for uncertainties and variabilities in parameter values and provide the technical basis for parameter ranges, probability distributions, or bounding values used in the performance assessment.
63.114(c)	Consider alternative conceptual models of features and processes that are consistent with available data and current scientific understanding, and evaluate the effects that alternative conceptual models have on the performance of the geologic potential repository.
63.114(d)	Consider only events that have at least one chance in 10,000 of occurring over 10,000 years.
63.114(e)	Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the performance assessment. Specific features, events, and processes of the geologic setting must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

Table 1.3-4. Proposed U.S. Nuclear Regulatory Commission Requirements for Performance Assessment (Continued)

Section of Proposed 10 CFR Part 63	Requirements
63.114(f)	Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.
63.114(g)	Provide the technical basis for models used in the performance assessment such as comparisons made with outputs of detailed process-level models and/or empirical observations (e.g., laboratory testing, field investigations, and natural analogs).
63.114(h)	Identify those design features of the engineered barrier system, and natural features of the geologic setting, that are considered barriers important to waste isolation.
63.114(i)	Describe the capability of barriers, identified as important to waste isolation, to isolate waste, taking into account uncertainties in characterizing and modeling the barriers.
63.114(j)	Provide the technical basis for the description of the capability of barriers, identified as important to waste isolation, to isolate waste.

Source: Proposed 10 CFR Part 63 (64 FR 8640 [101680])

Table 1.3-5. Proposed U.S. Nuclear Regulatory Commission Characteristics of the Reference Biosphere and Critical Group

Section of Proposed 10 CFR Part 63	**Characteristics**
63.115(a)	Reference biosphere.
63.115(a)(1)	Features, events, and processes that describe the reference biosphere shall be consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site.
63.115(a)(2)	Biosphere pathways shall be consistent with arid or semi-arid conditions.
63.115(a)(3)	Climate evolution shall be consistent with the geologic record of natural climate change in the region surrounding the Yucca Mountain site.
63.115(a)(4)	Evolution of the geologic setting shall be consistent with present knowledge of natural processes.
63.115(b)	Critical group.
63.115(b)(3)	The critical group resides within a farming community consisting of approximately 100 individuals, and exhibits behaviors or characteristics that will result in the highest expected annual doses.
63.115(b)(4)	The behaviors and characteristics of the average member of the critical group shall be based on the mean value of the critical group's variability range. The mean value shall not be unduly biased based on the extreme habits of a few individuals.
63.115(b)(5)	The average member of the critical group shall be an adult. Metabolic and physiological considerations shall be consistent with present knowledge of adults.

Source: Proposed 10 CFR Part 63 (64 FR 8640 [101680])

1.3.4 Proposed 10 CFR Part 963: Yucca Mountain Site Suitability Guidelines

In 1996, the DOE proposed (61 FR 66158 [100211]) to amend its general guidelines for site selection at 10 CFR Part 960 [126503], which it had promulgated under the Nuclear Waste Policy Act of 1982 [101681], 42 U.S.C. 10101 et seq. In 1998, the DOE issued the *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998 [101779]) as required by the Energy and Water Development Appropriations Act 1997, Public Law No. 104-206 [100008]. The report contains the bases for the site suitability criteria the DOE proposes to use and the methodology for applying the criteria to a design for a potential repository at the Yucca Mountain site. The current proposed rulemaking will limit 10 CFR Part 960 to preliminary site screening for repositories located elsewhere than Yucca Mountain and establish new suitability guidelines at proposed 10 CFR Part 963 (64 FR 67054 [124754]) for the Yucca Mountain site.

The proposed rule contains site suitability criteria and methods for considering the Yucca Mountain site for a potential nuclear waste repository. The suitability evaluation methods are consistent with the methods proposed by the NRC. The suitability criteria reflect current technical and scientific understanding and regulatory expectations (NRC and EPA) regarding the performance and safety of a potential geologic repository. These criteria are part of the program of scientific and technical investigations of the site to determine its natural properties and features (64 FR 67054 [124754]).

The DOE stated that "the phrase 'likely to meet applicable radiation protection standard' in [proposed] 10 CFR Part 963 is meant to clarify the role of the EPA standards and the NRC regulations in evaluating suitability and reaching a suitability determination" (64 FR 67054 [124754], p. 67075). The DOE has structured its rule regarding the methods and procedure for evaluating suitability to be consistent with proposed NRC licensing criteria and requirements. This is in recognition of NRC's broader role in the licensing process, and in anticipation of submitting an application for a license (64 FR 67054 [124754]).

The DOE's assessment of whether the Yucca Mountain site is suitable is a more preliminary assessment than the subsequent NRC licensing decision; hence, proposed 10 CFR Part 963 (64 FR 67054 [124754]) does not include all the NRC licensing requirements. The intent of proposed 10 CFR Part 963 (64 FR 67054 [124754]), as proposed, is to establish guidelines for providing the DOE with sufficient information to determine whether the site should be recommended to the President based on, among other things, whether the site is likely to meet applicable regulatory standards for licensing. The proposed guidelines do not address the entire process of site recommendation (Nuclear Waste Policy Act of 1982 [101681], Section 114; 64 FR 67054 [124754]).

1.3.4.1 Proposed Subpart A–General Provisions

The purpose of the proposed rule is to establish the methods and criteria for determining the suitability of the Yucca Mountain site for the location of a potential geologic repository. These methods and criteria will allow the DOE to analyze data from site characterization conducted under the Nuclear Waste Policy Act of 1982. Subpart A includes definitions of certain words and terms to clarify the DOE's intent and meaning and to make the terms consistent with proposed 10 CFR Part 63 (64 FR 8640 [101680]; 64 FR 67054 [124754]).

1.3.4.2 Proposed Subpart B–Site Suitability Determination, Methods, and Criteria

The scope of Subpart B includes the basis for the DOE's suitability determination for the Yucca Mountain site (64 FR 67054 [124754]). Subpart B is divided into two sections corresponding to the preclosure and postclosure periods, and each period is divided into three subsections. The subsections describe for each period: (1) the suitability determination; (2) the suitability evaluation method; and (3) the criteria to be used for the evaluation.

If the evaluation shows that the potential geologic repository is likely to satisfy the radiation protection standards for the preclosure and postclosure periods, then the DOE may determine that the site is suitable (64 FR 67054 [124754]). Tables 1.3-1 through 1.3-3 list the performance measures for the postclosure determinations. Table 1.3-6 contains the postclosure suitability evaluation method and Table 1.3-7 contains the postclosure suitability criteria for nondisruptive processes and events (see Table 1.3-2 and 1.3-3 for the criteria for disruptive processes and events). This method and the associated criteria prescribe how the DOE will demonstrate the long-term performance of the potential repository.

Table 1.3-6. Proposed U.S. Department of Energy Postclosure Suitability Evaluation Method

Section of Proposed 10 CFR Part 963	Total System Performance Assessment
963.16(b)	In conducting a [TSPA] under this section, DOE will:
963.16(b)(1)	Include data related to the suitability criteria in Sec 963.17
963.16(b)(2)	Account for uncertainties and variabilities in parameter values and provide the technical basis for parameter ranges, probability distributions, and bounding values
963.16(b)(3)	Consider alternative models of features and processes that are consistent with available data and current scientific understanding, and evaluate the effects that alternative models would have on the estimated performance of the geologic potential repository
963.16(b)(4)	Consider only events that have at least one chance in 10,000 of occurring over 10,000 years
963.16(b)(5)	Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting, including appropriate details as to magnitude and timing regarding any exclusions that would significantly change the expected annual dose
963.16(b)(6)	Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers, including those processes that would adversely affect natural barriers, (such as degradation of concrete liners affecting the pH of ground water or precipitation of minerals due to heat changing hydrologic processes), including appropriate details as to magnitude and timing regarding any exclusions that would significantly change the expected annual dose
963.16(b)(7)	Provide the technical basis for models used in the [TSPA] such as comparisons made with outputs of detailed process-level models and/or empirical observations (for example, laboratory testing, field investigations, and natural analogs)
963.16(b)(8)	Identify natural features of the geologic setting and design features of the engineered barrier system important to isolating radioactive waste
963.16(b)(9)	Describe the capability of the natural and engineered barriers important to isolating radioactive waste, taking into account uncertainties in characterizing and modeling such barriers

Table 1.3-6. Proposed U.S. Department of Energy Postclosure Suitability Evaluation Method (Continued)

Section of Proposed 10 CFR Part 963	Total System Performance Assessment
963.16(b)(10)	Provide the technical basis for the description of the capability of the natural and engineered barriers important to isolating radioactive waste
963.16(b)(11)	Use the reference biosphere and group receptor assumptions specified in applicable NRC regulations
963.16(b)(12)	Conduct appropriate sensitivity studies.

Source: Proposed 10 CFR Part 963 (64 FR 67054 [124754])

NOTE: See Tables 1.3-1, 1.3-2, and 1.3-3 for requirements of proposed 10 CFR 963.16(a) (64 FR 67054 [124754]).

Table 1.3-7. Proposed U.S. Department of Energy Postclosure Suitability Criteria–Nondisruptive Processes

Section of Proposed 10 CFR Part 963 and Characteristic	Criteria
963.17(a)	DOE will evaluate the postclosure suitability...through suitability criteria that reflect both the processes and the models used to simulate those processes that are important to the total system performance of the geologic potential repository. The applicable criteria are:
963.17(a)(1) Site	(i) Geologic properties of the site–e.g., stratigraphy, rock type and physical properties, and structural characteristics; (ii) Hydrologic properties of the site–e.g., porosity, permeability, moisture content, saturation, and potentiometric characteristics; (iii) Geophysical properties of the site–e.g., densities, velocities and water contents, as measured or deduced from geophysical logs; and (iv) Geochemical properties of the site–e.g., precipitation, dissolution characteristics, and sorption properties of mineral and rock surfaces;
963.17(a)(2) Unsaturated Zone Flow	(i) Climate–e.g., precipitation and postulated future climatic conditions; (ii) Infiltration–e.g., precipitation entering the mountain in excess of water returned to the atmosphere by evaporation and plant transpiration; (iii) Unsaturated zone flux–e.g., water movement through the pore spaces, or flowing along fractures or through perched water zones above the potential repository; and (iv) Seepage–e.g., water dripping into the...potential repository openings from the surrounding rock;
963.17(a)(3) Near Field Environment	(i) Thermal hydrology–e.g., effects of heat from the waste on water flow through the site, and the temperature and humidity at the engineered barriers, and (ii) Near field geochemical environment–e.g., the chemical reactions and products resulting from water contacting the waste and the engineered barrier materials;
963.17(a)(4) Engineered Barrier System Degradation	(i) Engineered barrier system component performance–e.g., drip shields, backfill, coatings, or chemical modifications, and (ii) Waste package degradation–e.g., corrosion of waste package materials in the near-field environment;
963.17(a)(5) Waste Form Degradation	(i) Cladding degradation–e.g., corrosion or break-down of the cladding on the individual spent fuel pellets; and (ii) Waste from dissolution–e.g., the ability of individual radionuclides to dissolve in water penetrating breached waste packages;

Table 1.3-7. Proposed U.S. Department of Energy Postclosure Suitability Criteria–Nondisruptive Processes (Continued)

Section of Proposed 10 CFR Part 963 and Characteristic	Criteria
963.17(a)(6) Engineered Barrier System Degradation, Flow, And Transport	(i) Colloid formation and stability–e.g., the formation of colloidal particles and the adherence of radionuclides to these particles as they may be washed through the remaining barriers; and (ii) Engineered barrier transport–e.g., the movement of radionuclides dissolved in water or adhering to colloidal particles to be transported through the remaining engineered barriers and in the underlying unsaturated zone;
963.17(a)(7) Unsaturated Zone Flow And Transport	(i) Unsaturated zone transport–e.g., the movement of water with dissolved radionuclides or colloidal particles through the unsaturated zone underlying the potential repository, including retardation mechanisms such as sorption on rock or mineral surfaces; (ii) Thermal hydrology–e.g., effects of heat from the waste on water flow through the site;
963.17(a)(8) Saturated Zone Flow And Transport	(i) Saturated zone transport–e.g., the movement of water with dissolved radionuclides or colloidal particles through the saturated zone underlying and beyond the potential repository, including retardation mechanisms such as sorption on rock or mineral surfaces (ii) Dilution–e.g., diffusion of radionuclides into pore spaces, dispersion of radionuclides along flow paths, and mixing with non-contaminated ground water;
963.17(a)(9) Biosphere	(i) Reference biosphere and receptor–e.g., biosphere water pathways, location and behavior of receptor (ii) Biosphere transport and uptake–e.g., the consumption of ground or surface waters through direct extraction or agriculture, including mixing with non-contaminated waters and exposure to contaminated agricultural products.

Source: Proposed 10 CFR Part 963 (64 FR 67054 [124754])

1.3.5 U.S. Nuclear Regulatory Commission Issue Resolution Status Reports

The DOE's site characterization program and resolution strategy are closely coordinated with the NRC. The NRC prelicensing program focuses on topics most critical to potential repository performance, termed Key Technical Issues (KTIs). These issues address technical matters regarding the performance of the site or the data needed to assess that performance (NRC 1999 [137163]). A goal of the site characterization program is consensus between the DOE and the NRC that the remaining KTIs have been addressed adequately or that adequate plans are in place to address the issues (64 FR 67054 [124754], p. 67062).

Through IRSRs, the NRC provides the DOE feedback on how to resolve the KTIs during the prelicensing consultation period. A resolved issue may be reopened if warranted (DOE 1998 [100548], p. 4-11). The NRC's next revision of the IRSRs, to be completed by the end of Fiscal Year 2000, will update information on progress in subissue resolution for each KTI (NRC 1999 [137163]). The KTIs focus NRC evaluations and foster an independent understanding of the issues and their relative importance to potential repository system performance. Various combinations of the KTIs include all of the principal factors that support the DOE's potential repository safety strategy (see Section 1.5.4.2). The KTIs all directly or indirectly relate to performance assessment (DOE 1998 [100550], p. 2-5). Table 1.3-8 shows the KTIs and their subissues.

Table 1.3-8. Subissues in U.S. Nuclear Regulatory Commission Key Technical Issues

KTI (IRSR rev. date)	KTI Subissue
USFIC Unsaturated and Saturated Flow Under Isothermal Conditions (NRC 1997 [100292]; NRC 1998 [102115]; NRC 1999 [140371])	USFIC1 Climate change
	USFIC2 Hydrologic effects of climate change
	USFIC3 Present-day shallow groundwater infiltration
	USFIC4 Deep percolation (present and future)
	USFIC5 Saturated zone ambient flow conditions and dilution processes
	USFIC6 Matrix diffusion
TEF Thermal Effects on Flow (NRC 1997 [100405]; NRC 1998 [102112]; NRC 1999 [137273])	TEF1 Sufficiency of thermal hydrologic testing program to assess thermal reflux in the near field
	TEF2 Sufficiency of thermal hydrologic modeling to predict the nature and bounds of thermal effects on flow in the near-field
	TEF3 Adequacy of TSPA with respect to thermal effects on flow
ENFE Evolution of the Near-Field Environment (NRC 1997 [100327]; NRC 1998 [102117]; NRC 1999 [105950])	ENFE1 Effects of coupled thermal-hydrologic chemical (THC) processes on seepage and flow
	ENFE2 Effects of coupled THC processes on waste package chemical environment
	ENFE3 Effects of coupled THC processes on chemical environment for radionuclide release
	ENFE4 Effects of THC processes on radionuclide transport through engineered and natural barriers
	ENFE5 Coupled THC processes affecting potential nuclear criticality in the near-field
CLST Container Life and Source Term (NRC 1998 [100410]; NRC 1998 [102114]; NRC 1999 [137277])	CLST1 Effects of corrosion on container lifetime and the release of radionuclidies to the near-field environment
	CLST2 Effects of materials stability and mechanical failure on container lifetime and the release of radionuclides to the near-field environment
	CLST3 Rate of degradation of SNF and the rate at which radionuclides in SNF are released to the near-field environment
	CLST4 Rate of degradation of HLW glass and the rate at which radionuclides in HLW glass are released to the near-field environment
	CLST5 Design of waste package and other components of the EBS for prevention of nuclear criticality
	CLST6 Effect of alternate design features on container lifetime and radionuclide release
RT Radionuclide Transport (NRC 1998 [102116]; NRC 1999 [136103])	RT1 Radionuclide transport through porous rock
	RT2 Radionuclide transport through alluvium
	RT3 Radionuclide transport through fractured rock
	RT4 Nuclear criticality in the far field
TSPAI Total System Performance Assessment and Integration (NRC 1998 [100296]; NRC 1998 [103760]; NRC 2000 [149372])	TSPAI1 System description and demonstration of multiple barriers
	TSPAI2 Scenario analysis within the TSPA methodology
	TSPAI3 Model abstraction within the TSPA methodology
	TSPAI4 Demonstration of the overall performance objective

Table 1.3-8. Subissues in U.S. Nuclear Regulatory Commission Key Technical Issues (Continued)

KTI (IRSR rev. date)	KTI Subissue
IA Igneous Activity (NRC 1998 [100297]; NRC 1998 [103603]; Reamer 1999 [119693])	IA1 Probability of future igneous activity
	IA2 Consequences of igneous activity within the potential repository setting
SDS Structural Deformation and Seismicity (NRC 1997 [100290]; NRC 1998 [101101]; NRC 1999 [135621])	SDS1 Faulting
	SDS2 Seismicity
	SDS3 Fracturing and structural framework of the geologic setting
	SDS4 Tectonics and crustal conditions
RDTME Repository Design and Thermal-Mechanical Effects (NRC 1997 [100404]; NRC 1998 [102113]; NRC 1999 [137163])	RDTME1 Implementation of an effective design control process within the overall quality assurance program
	RDTME2 Design of the geologic potential repository operations area for the effects of seismic events and direct fault disruption
	RDTME3 TM effects on underground facility design and performance
	RDTME4 Design and long-term contribution of potential repository seals in meeting pos-closure performance objectives

Sources: NRC 2000 [149372], App. B; DOE 1998 [100550], p. 2-5 ; NRC 2000 [151753]

NOTE: USFIC = Unsaturated and Saturated Flow under Isothermal Conditions; TEF = Thermal Effects on Flow; ENFE = Evolution of the Near-Field Environment; CLST = Container Life and Source Term; RT = Radionuclide Transport; TSPAI = Total System Performance Assessment and Integration; IA = Igneous Activity; SDS = Structural Deformation and Seismicity; RDTME = Repository Design and Thermal-Mechanical Effects

Each IRSR contains an (1) introduction, (2) definition of the KTI and all related subissues and the scope of the particular subissue(s) that is the subject of the IRSR, (3) importance of the particular subissue(s) to potential repository performance, (4) review methods and acceptance criteria, (5) status of resolution of the subissues, (6) references, and (7) an appendix summarizing those items resolved at the staff level and those items remaining open. The IRSR provides the technical basis for resolution of the subissues that will be used in subsequent reviews of the DOE submittals (NRC 1999 [137277], p. 2). Each IRSR is hierarchical, i.e., the IRSR identifies the subsystem affected (e.g., EBS), the primary issue (e.g., adequacy of EBS to provide long-term containment and limit releases), subissues (e.g., effects of corrosion processes on container lifetime), and components of subissues (e.g., humid-air corrosion and uniform aqueous corrosion).

1.3.5.1 Total System Performance Assessment and Integration Issue Resolution Status Report

Guidance for the NRC review of the TSPA is contained in the Total System Performance Assessment and Integration (TSPA&I) IRSR. The TSPA&I KTI describes an acceptable methodology for assessing potential repository performance and for using these assessments to demonstrate compliance with the overall performance objective and requirements for multiple barriers. Integration of information from many technical disciplines into the modeling and abstraction of the engineered system and natural FEPs is critical for an acceptable TSPA. The NRC included acceptance criteria for this integration in the TSPA&I IRSR to ensure that the transfer of information among the technical disciplines and to DOE's TSPA occurs, the analysis is focused on the integrated total system assessment, and the assessment is transparent, traceable, defensible, and comprehensive (NRC 2000 [149372], p. 3). The four TSPA&I subissues and their acceptance criteria describe the critical aspects of a TSPA methodology. These subissues and implications of their resolution follow.

System Description and Demonstration of Multiple Barriers–This subissue will ensure that the DOE has identified the design features of the EBS and natural features of the geologic setting that are considered important barriers to waste isolation, described the capability of the barriers important to waste isolation, and provided a technical basis for that description. It also ensures that compliance calculations in the TSPAs are clear and consistent and that the technical basis for the TSPA is sufficiently transparent and traceable.

Scenario Analysis–This subissue ensures that the TSPA appropriately considers likely processes and events by identifying, screening, and selecting the FEPs to be used in formulating scenarios in the TSPA. Guidance is provided on the construction of and assignment of probabilities to scenario classes, and their incorporation into an overall system performance.

Model Abstraction–This subissue ensures that the assumptions, conceptual approaches, data, models, and abstractions used in the TSPAs are appropriately integrated and technically defensible, and that technical support is commensurate with contribution to risk.

Demonstration of Overall Performance Objective–This subissue ensures the appropriate execution of the TSPA to demonstrate that the potential repository will satisfy the overall performance objectives under a range of FEPs. The objectives incorporate the standards to be set by the EPA at proposed 40 CFR 197 (64 FR 46976 [105065]) and adopted by the NRC in the final implementing rule, at proposed 10 CFR 63 (64 FR 8640 [101680]).

Because the TSPA&I IRSR addresses all subsystems, the issue hierarchy is somewhat different from that of the other IRSRs. The model abstraction subissue is further subdivided into integrated subissues of the potential repository system that must be appropriately abstracted into a TSPA (NRC 2000 [149372], p. 30). (In Revision 2 of the Total System Performance Assessment and Integration IRSR, the NRC replaced the term "KESA" (key elements of subsystem abstractions) with "ISI (integrated subissues)." These integrated subissues are related to the NRC KTIs as shown in Table 1.3-9. Figure 1.3-2 illustrates the hierarchy for the TSPA&I. The NRC "staff is currently developing a risk-informed and performance-based [Yucca Mountain] Review Plan for a potential [Yucca Mountain] potential repository LA based primarily on the Acceptance Criteria currently found in the *other* [emphasis added] KTI IRSRs" (NRC 2000 [149372], p. 2).

Table 1.3-9. Integrated Subissues for Nuclear Regulatory Commission Review of Model Abstraction within the Total System Performance Assessment Methodology Subsystem

Subsystem	Integrated Subissues	Related KTI Subissues
Engineered System	ENG1 Degradation of Engineered Barriers	TEF1,2 ENFE2 CLST1,2,6 RDTME3
	ENG2 Mechanical Disruption of Engineered Barriers	CLST1,2,5,6 IA2 SDS1-4 RDTME2,3
	ENG3 Quantity and Chemistry of Water Contacting the WPs and WFs	USFIC4 TEF1,2 ENFE1-3 CLST1,3,4,6 SDS3 RDTME3
	ENG4 Radionuclide Release Rates and Solubility Limits	ENFE3-5 CLST3-6

Table 1.3-9. Integrated Subissues for Nuclear Regulatory Commission Review of Model Abstraction within the Total System Performance Assessment Methodology Subsystem (Continued)

Subsystem	Integrated Subissues	Related KTI Subissues
Geosphere	UZ1 Spatial and Temporal Distribution of Flow	USFIC1,3,4 TEF1,2 ENFE1 SDS2,3 RDTME3
	UZ2 Flow Paths in the UZ	USFIC4 TEF1,2 ENFE1 SDS3
	UZ3 Radionuclide Transport in the UZ	USFIC4,6 ENFE4 RT1,3,4 SDS3
	SZ1 Flow Paths in the SZ	USFIC1,4,5 SDS3,4
	SZ2 Radionuclide Transport in the SZ	USFIC5,6 RT1-4 SDS3
	Direct1 Volcanic Disruption of Waste Packages	CLST1,2 IA1,2 SDS1,4
	Direct2 Airborne Transport of Radionuclides	IA2
Biosphere	Dose1 Dilution of Radionuclides in Groundwater due to Well Pumping	USFIC5
	Dose2 Redistribution of Radionuclides in Soil	IA2
	Dose3 Lifestyle of the Critical Group	IA2

Source: NRC 2000 [149372], p. 31

NOTE: Subissue 3–Integrated Subissues for U.S. Nuclear Regulatory Commission Review of Model Abstraction within the Total System Performance Assessment Methodology and Related KTI Subissues

See note in Table 1.3-8 for additional abbreviations WP = waste package; WF = waste form.

The NRC bases its judgment about which elements to abstract on "staff TSPAs performed in the past, review of the DOE's TSPAs, and knowledge of the design options for the [Yucca Mountain] site and [Yucca Mountain] site characteristics. Because TSPAs are considered iterative, some adjustment of the key elements may occur as future TSPAs and other relevant analyses are completed and site data are collected" (NRC 2000 [149372], p. 30). The NRC will review "elements of the DOE's total system performance demonstration and the relative contributions of potential repository subsystems or their components to identify those areas that require greater emphasis" (NRC 2000 [149372], p. 30). The NRC's completeness review will consider FEPs that could significantly impact performance. The adequacy review will consider how these FEPs are abstracted and integrated into the TSPA. The NRC will examine whether the engineered designs, site characteristics, and interactions among them have been appropriately identified, incorporated, and analyzed in the TSPA. The review will focus on understanding the importance to performance of the various assumptions, models, and input data in the TSPA and on ensuring that the degree of technical support for models and data abstractions is commensurate with contribution to risk.

1.3.5.2 Issue Resolution Status Report Treatment in Process Model Reports

Nine PMRs form the basis for the TSPA. These reports summarize the data, assumptions, and analyses documented in detail in subsidiary analysis and model reports.

- *Integrated Site Model Process Model Report* (CRWMS M&O 2000 [146988])
- *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774])

- EBS: *Engineered Barrier System Degradation, Flow, and Transport Process Model Report* (CRWMS M&O 2000 [145796])

- WP: *Waste Package Degradation Process Model Report* (CRWMS M&O 2000 [138396])

- WF: *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332])

- NFE: *Near Field Environment Process Model Report (PMR)* (CRWMS M&O 2000 [153178])

- SZ: *Saturated Zone Flow and Transport Process Model Report* (CRWMS M&O 2000 [145738])

- Bio: *Biosphere Process Model Report* (CRWMS M&O 2000 [151615])

- DE: *Disruptive Events Process Model Report* (CRWMS M&O 2000 [141733]).

Table 1.3-10 correlates the NRC TSPA&I Integrated Subissues with the DOE PMRs. Refer to tables D.1-4 and D.1-5 of Appendix D, *Issue Resolution Status Reports Tracking Database,* of this report for details of the approach used in the relevant PMRs for each of the TSPA&I Integrated Subissues, and for a crosswalk of the KTIs to the PMRs.

Table 1.3-10. Correlation Between the U.S. Nuclear Regulatory Commission's Total System Performance Assessment and Integration Integrated Subissues and the U.S. Department of Energy's Process Model Reports

TSPA&I ISIs	DOE Process Model Reports								
	ISM	Bio	DE	EBS	NFE	SZ	UZ	WF	WP
ENG1									
ENG2									
ENG3									
ENG4									
UZ1									
UZ2									
UZ3									
SZ1									
SZ2									
DIRECT1									
DIRECT2									
DOSE1									
DOSE2									
DOSE3									

1.4 PHILOSOPHY OF TOTAL SYSTEM PERFORMANCE ASSESSMENT

The TSPA has become the internationally recognized method for analyzing system behavior for nuclear waste repositories. It is important to understand why TSPAs are performed, the unique nature of a TSPA compared to other types of analyses, and why the confidence in TSPA as a process has been established at such a global level.

1.4.1 Why Total System Performance Assessments Are Performed

Performance assessments are used to forecast how a specific system and all of its components evolve over time. Comparing the results to performance requirements allows analysts to estimate whether the amount of harmful material that may become accessible in the environment is acceptably low. The requirements, usually in the form of regulatory criteria, are generally established by governmental oversight agencies. The ultimate determination of whether a system complies with the requirements lies with the legally responsible regulatory group. The task of proposing a nuclear waste repository is to provide reasonable assurance that the safety standard will be met, which, in turn, requires that:

- The proposed system and all of its components are understood.
- The capability to model the system can be demonstrated.
- The uncertainties in the analysis can be adequately accounted for and treated.
- The information in the model provides reasonable assurance that safety standards will be met.

In addition to providing a tool for determining whether a system meets regulatory requirements, TSPA also provides a rigorous method for aiding management in establishing the priority of information-gathering activities during the site selection, site characterization, and design phases. As more information is gathered, the TSPA process iterates to incorporate revised and updated information into successive TSPA models. This allows the program to progress toward more reasonable and defensible total system models. Results of each TSPA, particularly the sensitivity and uncertainty studies, provide information about the relative importance of ongoing or proposed information-gathering activities addressing site characterization and design development. Successive TSPAs require that the total system models become more representative. Several TSPAs have been completed on the proposed Yucca Mountain repository system (Sinnock et al. 1984 [100553]; Barnard and Dockery 1991 [100307]; Barnard et al. 1992 [100309]; Eslinger et al. 1993 [100554]; Wilson et al. 1994 [100191]; CRWMS M&O 1994 [100111]; CRWMS M&O 1998 [108000]). These efforts, along with studies done by other organizations (Wescott et al. 1995 [100476]; Kessler et al. 1996 100558]), have contributed to the iterative process of the TSPA for the SR. The progression of these analyses is described in Section 1.5.

A TSPA is unique in that the analysis links all the system components together. This linkage is important because it allows each component to be viewed in the context of the behavior of the entire system. Even the simplest system has various aspects that are easier to understand when studied separately (e.g., waste package material degradation may be characterized by laboratory tests of corrosion). However, the geologic system in which the waste package is to be emplaced may be analyzed using field studies of the host rock for properties that are only observed on a large scale (e.g., fracture density), as well as laboratory studies of other aspects (e.g., water chemistry). In a functioning system, these elements provide feedback to one another. The influence of thermal output from the waste on the water chemistry in the near-field could lead to altered corrosion of the waste packages. This very simple example shows an obvious potential for feedback. When the numerous components of a very complex system are brought together and simulated as a single, integrated system in a TSPA, interactions among the components that would not otherwise be identified in a single component analysis may be identified in the TSPA analysis.

The proposed repository safety strategy for the Yucca Mountain Site Characterization Project (YMP) (CRWMS M&O 2000 [148713]) relies on a multiple barrier system. This isolation strategy means that the components of the natural and engineered systems form a series of barriers. Because the behavior of each component in the series is governed by a different set of physical or chemical processes, this strategy provides a strong argument that the entire system is very unlikely to fail in response to a single mechanism. Also, the use of different types of barriers precludes reliance on complete knowledge about any one process. Therefore, the incorporation of multiple barriers helps to answer the question that frequently arises (i.e., How can the analysis account for what is not known?). Given the uncertainty inherent in a forecast, one way to deal with an unanticipated response by one component of the system is to have multiple additional components that will continue to operate as barriers in the face of the unanticipated response.

The concept of reasonable assurance used by the NRC in its proposed regulations for the potential Yucca Mountain repository does not require absolute certainty for the results of an analysis. The incorporation of uncertainty into the TSPA, using various mathematical methods, allows the regulator and others to determine if the goal of reasonable assurance has been met. (See Section 5 for the study of uncertainty.) However, some of the general methods of treating uncertainty include developing distributions to represent various types of data and assigning probabilities to different conceptual models to encompass a range of potential behaviors (or responses) of certain components.

1.4.2 Why Total System Performance Assessments Are the Appropriate Tool for Analyzing the Safety of Repository Systems

A question that often arises is whether or not performance assessment is a useful tool for the purpose of analyzing safety. The consensus of the international waste management community is that, in the realm of providing reasonable assurance, performance assessment is an adequate tool. In support of this consensus, the Nuclear Energy Agency (NEA) Radioactive Waste Management Committee and the International Atomic Energy Agency International Radioactive Waste Management Advisory Committee issued a collective opinion that they…

> Confirm that safety assessments are available today to evaluate adequately the potential long-term radiological impacts of a carefully designed radioactive waste disposal system on humans and the environment, and consider that appropriate

use of safety assessment methods, coupled with sufficient information from proposed disposal sites, can provide the technical basis to decide whether specific disposal systems would offer to society a satisfactory level of safety for both current and future generations (Nuclear Energy Agency 1991 [100477], p. 7).

Although TSPAs can never be proven to be absolutely valid, many environmental problems require modeling of long-term interactions of man-made and geologic systems. Using the term model acknowledges that whether or not the descriptions of geologic FEPs are unique and represent absolute reality will never be known. Validation of a long-term predictive model means that, on the basis of tests of the assumptions, inputs, outputs, and sensitivities, the model adequately reflects the recognized behavior of the portion of the system it is intended to represent. Adequacy is driven by the needs of the application for which the model is developed (Boak and Dockery 1998 [100368], pp. 178 to 180).

Scientists assessing long-term risk use the following mechanisms to establish the adequacy of their models (Boak and Dockery 1998 [100368], pp. 181 to 182):

- Conservatism–in assigning parameter values and process descriptions, including ignoring some potentially mitigating processes
- Stochastic simulation–to assess the effect of uncertainty in descriptions and the sensitivity of performance predictions to uncertainty and to examine alternative scenarios and process models
- Expert judgment–to assign appropriate ranges of parameters where data are sparse, controversial, or unobtainable.

Measures undertaken to demonstrate that the effort to ensure adequacy has been comprehensive include (1) documentation of the model structures, including justification for assumptions and simplifications, as well as the examination of alternative conceptualizations for the system, and (2) review by the scientific community and those who have a stake in the decisions that these models support (Boak and Dockery 1998 [100368], p. 182).

Uncertainty is an inherent part of all total system studies. Information-gathering activities are directed at reducing uncertainty as much as is practical. However, because of natural variability in the systems being studied and limited understanding about how processes will operate in the future, uncertainty will always have to be explicitly included in TSPA calculations.

1.4.3 Evaluating Confidence

Evaluating confidence in the TSPA requires a combination of the efforts to demonstrate adequacy and to evaluate uncertainties. A case needs to be built for the defensible basis of reasonable assurance that the long term impacts are either reasonably or conservatively evaluated, and clearly and traceably documented.

To provide a statement of confidence, several assurances must be given in proper documentation. Assurances must be provided of the systematic, and arguably complete nature of the FEPs identification and selection approaches. Assurances must be provided that the selection of FEPs, and the exclusion of FEPs, has been carefully and correctly done in a systematic, traceable way. There needs to be a clear path from data to process models and finally to abstractions of models used in the TSPA model itself.

Then the TSPA, and its internal and external linkages need to be shown to be properly based in science, and not arbitrary. Finally, the aforementioned uncertainty and sensitivity analyses need to be used, and thus need to be selected to be useful for, demonstrating the value of information from data and models, providing an ability to make risk-informed, performance-based findings of fact based on TSPA results. Utilization of this approach provides confidence in the analyses presented herein.

1.5 PREVIOUS U.S. DEPARTMENT OF ENERGY TOTAL SYSTEM PERFORMANCE ASSESSMENTS FOR THE YUCCA MOUNTAIN SITE

TSPA provides a tool for evaluating the significance of uncertainties in properties and processes by evaluating the sensitivity of calculated results to different assumptions about these properties and processes. The site characterization process proceeds iteratively (i.e., new data and design changes are incorporated into updated TSPA models, and updated TSPA sensitivity studies suggest where new data and design enhancements might be valuable). Over time, this iterative process reduces uncertainty in the forecasted performance of the potential repository. The TSPA tool has been used by the DOE to identify needs and to set priorities for site characterization work, materials-properties investigations, and other design-related investigations, including repository and waste package design options. Figure 1.5-1 illustrates the major TSPA iterations.

The DOE conducted benchmark performance assessments of the total potential repository system in 1991, 1993, 1995, and 1998. As a result, the DOE TSPA has become much more sophisticated and representative. The tool and the analyses have evolved as new data have been collected. The DOE has continually added component models that provide more details of system behavior. The abstractions of the components of the system have become more sophisticated with each DOE TSPA. This is described in detail in this section, which briefly describes the purpose, objectives, and goals of each TSPA, identifies the primary issues investigated, and summarizes the conclusions reached.

1.5.1 Total System Performance Assessment-1991

In 1991, the DOE conducted the *TSPA-1991: An Initial Total-System Performance Assessment for Yucca Mountain* (Barnard et al. 1992 [100309]). Sandia National Laboratories performed the TSPA for the YMP.

1.5.1.1 Purpose

The primary purpose of the 1991 TSPA effort was the derivation of abstracted representations capturing the essence of the complex processes that contribute to the behavior of a repository system. Defensible, abstracted representations were deemed to be necessary tools for producing useful estimates of the principal performance measure for evaluating compliance with the EPA standard, 40 CFR Part 191 (50 FR 38066 [100495]). A secondary purpose was to demonstrate that complex combinations of probabilistic data could be assembled to provide a reasonable overall estimate of system performance. Because of the limited number of components included, this performance estimate did not evaluate the suitability of Yucca Mountain as a site for a potential radioactive waste repository. The results were intended to provide guidance for site characterization and for the next iterations of TSPA (Barnard et al. 1992 [100309], p. 1-2).

1.5.1.2 Conclusions

TSPA-1991 showed that complex processes can be abstracted into more simplified representations that retain the necessary degree of sensitivity, consistent with the understanding of the processes and work done using other models and techniques. As prescribed in 40 CFR Part 191 (50 FR 38066 [100495]), the results were combined into a conditional, total-system, complementary, cumulative-distribution function, thereby demonstrating that total system performance can be estimated (Barnard et al. 1992 [100309], p. 10-1).

The results reflected considerable uncertainty and many conservative assumptions, which were attributed to the lack of site-specific data. Uncertainty in the models was partially addressed by using two alternative conceptual models of flow in the UZ. The calculated releases were sensitive to the choice of flow model. Because a sensitivity study was not done, the most important parameters for nominal conditions were not identified. Because of the uncertainty and conservatism, the analyses were not an appropriate basis for site suitability recommendations (Barnard et al. 1992 [100309], p. 10-1). Figure 1.5-2 illustrates the subsystem component model abstractions that were available for the 1991 TSPA. The analyses were considered adequate for guiding site characterization activities. The 1991 TSPA provided the following general recommendations for future work:

- Develop an exhaustive set of scenario categories
- Select a formal method for future calculations
- Continue to validate the abstractions used in the TSPA
- Develop and integrate new alternative conceptual models
- Use TSPA analyses to help guide site characterization
- Analyze new site characterization data for incorporation into TSPA analyses
- Investigate the effects of disturbing conditions
- Investigate general thermal effects caused by potential repository heating.

Recommendations for five components of the analyses were also provided (see Table 1.5-1). These components were parameters, aqueous flow and transport, gaseous flow and transport, human intrusion, and basaltic igneous activity (Barnard et al. 1992 [100309], p. 10-1).

Table 1.5-1. Parameter-Specific Recommendations for Future Work from the 1991 Total System Performance Assessment

Components	Recommendations
Data Set	Develop alternative interpretations of the Yucca Mountain geohydrologic stratigraphy
	Conduct a formal sensitivity study to identify the most important parameters
	Refine further the elicitation techniques employed to develop the data set
	Refine parameter distributions, as additional information becomes available
	Analyze correlation among parameters

Table 1.5-1. Parameter-Specific Recommendations for Future Work from the 1991 Total System Performance Assessment

Components	Recommendations
Data Set (Continued)	Develop hydrogeologic and geochemical parameter values for the SZ
	Quantify the effects of scale on the model parameters
	Investigate the effects of heterogeneity among the stratigraphic units
	Investigate the validity of the one-dimensional modeling
Source Term	Develop more accurate, correlated, and defensible parameter distributions
	Perform aqueous-transport analyses, including all significant radionuclides
	Include the waste container and the fuel-rod cladding more realistically
	Develop submodels for additional release modes
	Verify the validity of the alteration-limited-release model
Geochemistry	Refine parameter distributions
	Develop retardation information for all significant radionuclides
	Include retardation for transport in fractures, if it can be shown to be significant
	Study the effects of colloids, especially of plutonium and americium
	Investigate methods for modeling radionuclide transport other than K_d values
Aqueous Flow and Transport in the UZ	Study and include the effects of spatial correlation
	Refine the weeps model
	Study climate change and its effects on percolation flux
	Investigate effects of repository heating on groundwater flow and transport
	Verify the conceptual models in the TSPA
Aqueous Flow and Transport in the SZ	Improve the coupling of the UZ and the SZ and account for the uncertainty in travel time
	Investigate effects of matrix and fracture coupling
	Investigate effects of seismic, tectonic, and volcanic activity
Gaseous Flow and Transport	Calculate aqueous and gaseous releases together
	Characterize gas permeability throughout the UZ
	Include the uncertainty in the permeabilities of welded and nonwelded tuff
	Calculate the travel-time distributions for the potential repository-temperature curve realistically
	Characterize the ^{14}C inventory, prompt fraction, and release rate
	Calculate travel-time distributions with a model that couples gas flow and thermal effects
	Continue to investigate carbon geochemistry and rock interactions
Human Intrusion	Determine the likelihood that commercially attractive natural resources are present
	Complete the human-intrusion event tree
Basaltic Igneous Activity	Review the complete event tree for igneous events and estimate probabilities for igneous activity
	Consider nonmechanical interactions between magma and waste
	Investigate the interaction depth for wall-rock erosion
	Obtain a better understanding of the depth at which vesiculation occurs

Source: Barnard et al. 1992 [100309], Chapter 11

1.5.2 Total System Performance Assessment-1993

Beginning in fiscal year 1993, the Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) was assigned the responsibility to plan, coordinate, and contribute to the second iteration of TSPA *Total System Performance Assessment-1993: An Evaluation of the Potential Yucca Mountain Repository* (CRWMS M&O 1994 [100111], p. 1-2).

Both the CRWMS M&O and SNL (Wilson et al. 1994 [100191]) conducted TSPA analyses for the 1993 iteration. The CRWMS M&O work is summarized here. Although they used two different computational tools for assessing the total system performance, the primary difference in the two approaches was the level of detail incorporated into the abstraction from the process models. The assumptions made in the two sets of analyses were much more important to the results than was the computational scheme embodied in the codes used. Subsequently, the computational scheme used by the CRWMS M&O was selected for future TSPA analyses.

1.5.2.1 Goals and Objectives

The goals of the CRWMS M&O in TSPA-1993 were to (1) enhance the realism and representativeness of the analyses, (2) incorporate new information and designs that had become available since the completion of TSPA-1991, (3) test the sensitivity of the predicted performance against various conceptual model-and-parameter uncertainties, and (4) evaluate alternate measures of postclosure performance (CRWMS M&O 1994 [100111], p. 1-2). The analyses, aimed at identifying the key assumptions and the sensitivity of the results to those assumptions, had eight major objectives (CRWMS M&O 1994 [100111], p. 1-4).

- Incorporate thermal dependency on individual processes and parameters
- Evaluate the effects of alternate thermal loads
- Evaluate the effects of alternate waste package designs
- Evaluate alternate measures of total system performance
- Incorporate new site and design information
- Incorporate a more representative inventory, including high-level waste
- Conduct sensitivity analyses to identify the key processes and parameters
- Provide guidance to site characterization and design activities.

1.5.2.2 Primary Issues

At the time of TSPA-1993, the National Academy of Sciences (NAS) was evaluating the appropriateness of a dose-based standard for Yucca Mountain (Section 1.3.1.1). Therefore, the performance measure was itself a primary issue. Another issue arising from the lack of a standard was the time period of regulatory concern. (CRWMS M&O 1994 [100111], p. 1-2)

The ability to forecast performance of the site and engineered barriers in containing and isolating radioactive wastes from the accessible environment depended on two primary issues. These were the understanding of flow and transport through the fractured-porous media and the uncertainty in the magnitude of the percolation flux through the UZ (CRWMS M&O 1994 [100111], p. 4-14). The primary design issues affecting the migration of radionuclides away from the disposal area were the thermal load and the waste package design, and their effects on aqueous corrosion of the waste form (CRWMS M&O 1994 [100111], p. 4-12).

1.5.2.3 Conclusions

Although the TSPA-1993 (CRWMS M&O 1994 [100111]) analyses significantly extended the work performed in TSPA-1991 (Barnard et al. 1992 [100309]), uncertainties remained. The sources of significant uncertainty in TSPA-1993 and the associated recommendations for future work (CRWMS M&O 1994 [100111]) are given in Table 1.5-2. Figure 1.5-3 illustrates the subsystem component model abstractions that were available for the 1993 TSPA. Many of these recommendations were also made in the SNL TSPA for 1993. The following additional recommendations were made by SNL (Wilson et al. 1994 [100191], p. ES-21).

- Characterize the spatial distribution of bulk permeability
- Collect data on adsorption of CO_2 to tuff
- Characterize the horizontal and vertical dispersion in the SZ
- Evaluate heterogeneity and spatial correlation for geostatistical modeling
- Evaluate cross-correlation among parameters
- Determine thermal and hydraulic properties of proposed backfill materials
- Characterize fault-zone hydrogeologic properties.

Table 1.5-2. Sources of Uncertainty and Recommendations from Total System Performance Assessment-1993

Sources of Significant Uncertainty	Recommendations
Panel and drift scale thermo-hydrologic analyses	Perform additional analyses to evaluate the effect of uncertain and spatially variable thermo-hydrologic properties, uncertain fracture-matrix conceptual models, and uncertain ambient percolation fluxes on the expected far-field, near-field, and very-near-field (waste package scale) thermal and hydrologic regimes as a function of space and time.
Initiation and rates of aqueous corrosion processes	Develop a greater understanding of the cathodic protection of the inner container, the processes affecting pitting, and even the definition of waste package failure, in order to provide a more defensible argument for the range of likely waste package lifetimes.
Ambient UZ percolation flux	Employ direct or indirect observations to better quantify the expected UZ percolation flux and its uncertainty.
Fracture-matrix interactions as water moves through the UZ	Incorporate the preliminary, site-scale, UZ model to be completed by the U.S. Geological Survey in fiscal year 1994.
	Conduct further testing to determine the relative significance of alternate conceptualizations to the composite porosity model of fracture-matrix interaction.
	Validate the simplified K_d representation of radionuclide transport through the UZ.

Source: CRWMS M&O 1994 [100111], pp. 4-17 to 4-18

Sandia National Laboratory also concluded that regulatory change could lead to significant changes in program priorities for site characterization. A performance measure based on individual dose for the time period of regulatory concern, as in proposed 40 CFR Part 191 (58 FR 66398 [107802]), would require additional characterization of the biosphere. A longer time period would lead to more emphasis on determining radionuclide release rates (Wilson et al. 1994 [100191], p. ES-22).

1.5.3 Total System Performance Assessment-1995

The work performed across the gamut of the YMP had been melded into a single TSPA effort by the time the *Total System Performance Assessment-1995: An Evaluation of the Potential Yucca*

Mountain Repository (CRWMS M&O 1995 [100198]) was initiated. Four specific goals were identified for the 1995 iteration of TSPA (CRWMS M&O 1995 [100198], pp. 1-5 to 1-7).

1. Utilize what were believed to be more representative conceptual models that built upon the assumptions employed in TSPA-1993, in particular, for the treatment of the EBS, including the waste package, using reasonably conservative representations of the relevant processes and parameters affecting total system performance.

2. Incorporate more recent design information than was available for TSPA-1993, evaluating a range of alternative conceptual models and parameters to explicitly address the uncertainty and variability in the understanding and the significance of that uncertainty on the predicted performance.

3. Utilize the most recent site information and models, acknowledging their uncertainty and variability, focusing the analyses on those components of the waste containment and isolation system that are most sensitive.

4. Evaluate the EBS release performance measure, as well as alternative measures of total system performance, using a range of possible measures of safety, including cumulative radionuclide releases, peak concentrations, or doses.

The focus of the 1995 TSPA was on those components of the system that were determined in the 1993 TSPA to be most significant in containing and isolating radioactive wastes from the biosphere. These were the engineered components of the system and the near-field environment in which the engineered components reside (CRWMS M&O 1995 [100198], p. 1-3).

1.5.3.1 Primary Issues

As the EPA had not yet proposed an environmental standard for Yucca Mountain (Section 1.3.2), the performance measure remained an issue. Technical issues (CRWMS M&O 1995 [100198], pp. 10-1 to 10-2) were:

- Alternative models of the thermo-hydrologic environment near the waste package
- Alternative assumptions about the degradation of the waste package materials
- Alternative assumptions about capillary barriers in the drifts
- Alternative concepts of advective flow in the drifts and percolation flux in the UZ
- Alternative conceptual models of transport in the UZ
- Alternative thermal-loading designs and backfill emplacement options.

1.5.3.2 Conclusions

Five different measures of performance were evaluated in the 1995 TSPA (CRWMS M&O 1995 [100198]). The first two considered subsystems: the waste package (substantially complete containment) and the EBS (the peak radionuclide release rate). The remaining three measures quantified total system performance: the cumulative radionuclide release at the accessible environment over 10,000 years; and the maximum radiation doses in both 10,000 and 1 million years to an individual located at the accessible environment boundary. Table 1.5-3 presents the factors determined to affect performance in the 1995 TSPA. Figure 1.5-4 illustrates the subsystem component model abstractions that were available for the 1995 TSPA. The TSPA team identified a detailed technical analysis of the robustness of the process models under development as the performance assessment activity having the highest priority in preparation for the next full iteration of TSPA. Equally significant was assuring that the developed and substantiated process models could be appropriately abstracted for use in the next TSPA (CRWMS M&O 1995 [100198], p. 10-26). The necessary process models are shown in Table 1.5-4.

Table 1.5-3. Factors Affecting Performance in the 1995 Total System Performance Assessment

Performance Measure	Factor Affecting Performance
Substantially Complete Containment at the Waste Package	The rate of container degradation was not directly correlated with the thermal load, given the assumptions (validity of assumptions needed substantiation).
	Incorporating cathodic protection significantly extends the lifetime of the waste packages (sensitivity analyses with unconfirmed, first-order approximations).
	Incorporating time dependence for pitting of corrosion-resistant materials significantly affects the predicted failure distribution (sensitivity analyses in lieu of an improved, experimentally derived model).
	Conceptual representations of drift scale thermal-hydrology and corrosion-degradation models significantly affect the waste package failure distribution over the first 10,000 years but are less significant for longer times.
Peak Release Rate from the EBS	The conceptualization of diffusion resulted in very small diffusive releases (drip rate required substantiation).
	When advection dominated the EBS release, the infiltration-rate distribution had a significant effect and the conceptualization (e.g., assuming a capillary barrier) of how dripping water contacts the waste package was important.
	The mode of radionuclide transport, gas phase or dissolved in liquid, affected the peak EBS release rates and peak doses at the accessible environment.
	The dissolution rate did not significantly affect the peak EBS release rate.
Cumulative Release of Radionuclides at the Accessible Environment–10,000 years	Certain conceptual assumptions resulted in engineered barriers that provided complete containment and natural barriers that provided complete isolation.
	The most conservative assumptions for both EBS and natural-barrier performance resulted in the system being dominated by the percolation flux distribution (affecting the likelihood of dripping, the magnitude of the advective release from the EBS, and the distribution of radionuclide transport and matrix velocity through the UZ).
Peak Radiation Dose to Reasonably Maximally Exposed Individual at the Accessible Environment–10,000 Years	Factors that delayed the arrival of the peak concentration of radionuclides at the accessible environment were significant, primarily the percolation flux distribution, but sorption, matrix diffusion, and fracture-flow path length also affected arrival time.
	Predicted peak arrival time generally occurred between 10,000 and 1 million years, depending on the nuclide and the flow and transport conceptualization.
	Dispersion in the UZ reduced the arrival time and increased the peak dose during the 10,000-year time period.
	Dilution of radionuclide concentrations in the SZ, dependent on the local Darcy flux distribution within the SZ, controlled both peak concentrations and peak doses, but not cumulative releases.

Table 1.5-3. Factors Affecting Performance in the 1995 Total System Performance Assessment (Continued)

Performance Measure	Factor Affecting Performance
Peak Radiation Dose to Reasonably Maximally Exposed Individual at the Accessible Environment–1 million Years	Factors that delayed the arrival of the peak concentration at the accessible environment were less significant because of the extremely long time period and the long half-lives of some key radionuclides.
	Waste package and site performance helped contain and isolate radioactive wastes, but were unlikely to preclude the release of ^{99}Tc, ^{237}Np, and ^{129}I over a 1-million-year time period.
	Dispersion and dilution in the geosphere were significant processes in reducing peak concentrations and peak doses.
	Diffusion-dominated releases from the EBS significantly reduce the peak release rate with either a very low percolation flux distribution or an efficient capillary barrier.

Source: CRWMS M&O 1995 [100198], pp. 10-3 to 10-8

Table 1.5-4. Process Models Necessary for Development of Abstractions Beyond Those in the 1995 Total System Performance Assessment

Priority	Process Model	Notes
1	Site scale UZ hydrology model(s) (ambient)	UZ: unsaturated zone
3	Repository scale UZ TH model(s)	TH: thermal-hydrology
3	Site scale UZ geochemical model(s) (ambient)	
1	Drift scale TH model(s)	
3	Drift scale TC model(s)	TC: thermal-chemical
4	Drift scale TM model(s)	TM: thermal-mechanical (potentially higher priority, if no backfill in drift)
4	Drift-scale-coupled THCM model(s)	THCM: thermal-hydrological-chemical-mechanical
2	Waste package degradation model(s)	
4	Cladding degradation model(s)	
3	Waste form dissolution model(s)	
2	Waste package scale TC model(s) (solubility)	
2	Drift scale transport model(s)	
3	Site scale UZ transport model(s)	
3	Regional and site scale SZ flow model(s)	SZ: saturated zone
3	Site scale SZ transport model(s)	
3	Biosphere transport model(s)	(Because no standard had been promulgated, the recommendation was that the EPA should prescribe model for dose or risk standard)
4	Tectonics direct and indirect effects model(s)	
3	Volcanic direct and indirect effects model(s)	
2	Climate change indirect effects model(s)	

Source: CRWMS M&O 1995 [100198], p. 10-27

1.5.4 Total System Performance Assessment-1998

The *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998 [101779]), proposed by the DOE in 1996 and mandated by Congress in 1997, was designed to provide the President, Congress, and the public with information on the progress of the YMP. The assessment also identified the critical issues that should be addressed before a decision can be made by the Secretary of Energy on whether to recommend the Yucca Mountain site for a potential repository. The keystone of the viability assessment was the TSPA, documented in Volume 3 (DOE 1998 [100550]).

1.5.4.1 Goals

The statutory goal of the viability assessment TSPA was to describe the probable behavior, relative to the overall system performance standards, of a potential repository in the Yucca Mountain geologic setting, based on the design concept and the scientific data and analyses available by 1998 (DOE 1998 [100547], p. 1). The DOE also wanted to assess quantitatively the total system performance so that the significance of each of the key components in the potential repository safety strategy could be defined to assist in a systematic refocusing of the project resources.

This was accomplished by examining the relative importance of the various TSPA components and parameters through sensitivity and uncertainty analyses. The information about uncertainty assisted the DOE in defining the work required either to reduce uncertainty or to modify the potential repository design to accommodate this uncertainty before proceeding with the Site Recommendation. The TSPA also provided a vehicle for prelicensing discussions with the NRC. An important role of the TSPA was to evaluate the potential significance of KTIs identified by the NRC (Section 1.3.5.1). Another goal was to produce a document that transparently described for all interested parties the assumptions, the intermediate steps, the results, and the conclusions of the analyses (DOE 1998 [100550], p. 2-3).

1.5.4.2 Primary Issues

The primary issues were the KTIs that the NRC considered most important to potential repository performance. These were:

- Total System Performance Assessment and Integration
- Unsaturated and Saturated Flow under Isothermal Conditions
- Evolution of the Near-Field Environment
- Container Life and Source Term
- Repository Design and Thermal Mechanical Effects
- Thermal Effects on Flow
- Radionuclide Transport
- Structural Deformation and Seismicity
- Igneous Activity.

All relate to performance assessment (DOE 1998 [100550], Volume 3, p. 2-5). As was true for the preceding TSPAs, the EPA had not yet proposed an environmental standard for Yucca Mountain. Consequently, the NRC had not proposed revising 10 CFR Part 60, which governed such repositories (Section 1.3.1.1).

1.5.4.3 Conclusions

By the time of this TSPA iteration, the DOE had developed a potential repository safety strategy (DOE 1998 [100550], Volume 3) having four key attributes for system performance. The attributes are (1) limited water contacting waste packages, (2) long waste package lifetime, (3) low rate of release of radionuclides from breached waste packages, and (4) radionuclide concentration reduction during transport from the waste packages. The YMP had identified nineteen principal factors for analyzing system performance and developed modeling components to examine the factors.

The probable behavior of the reference design suggested that the vast majority of radionuclides in the waste are immobile and never leave the potential repository, even if in contact with water. A few radionuclides appeared sufficiently mobile under some conditions that they could reach the biosphere downgradient from the potential repository. Hence, the most important factors for system performance over time were the amount of water likely to contact the waste packages and the amount of waste exposed to that water. Consequently, factors that limit the contact of water with the waste were considered highly important to performance. Under the base-case scenario, the quantities of radionuclides reaching the biosphere were small: a negligible amount in 10,000 years and a dose rate for hundreds of thousands of years that is comparable to natural background activity (DOE 1998 [100550], Volume 3, p. 6-5). Figure 1.5-5 illustrates the subsystem component model abstractions that were available for the viability assessment TSPA and identifies remaining uncertainties.

Table 1.5-5 shows the four key attributes of the DOE strategy for repository safety and the associated nineteen principal factors as addressed by the components of the TSPA. This table relates the key attributes, the principal factors, and the model components to the corresponding issues identified by the NRC. The sensitivity of the results to uncertainties in the estimates is also summarized for three periods of performance. High significance means that uncertainty in the principal factor, or its absence, results in a factor of over 50 increase (or decrease) in peak dose rate from the expected value. Medium significance means a factor of 5 to 50 increase (or decrease), and low significance means less than a factor of 5 increase (or decrease) in peak dose rate from the expected value. These indicators of sensitivity guided additional work. (DOE 1998 [100548], Volume 3, p. 6-12).

Table 1.5-5. Total System Performance Assessment-Viability Assessment: Attributes and Principal Factors with Significance of Uncertainty, Model Components, and Key Technical Issues

Attributes of the Potential Repository Safety Strategy	Principal Factor	Significance of Uncertainty by Performance Period (x 1,000 years)			TSPA Model Component	U.S. Nuclear Regulatory Commission Key Technical Issue
		10	100	1,000		
Limited water contacting waste packages	Precipitation and infiltration of water into the mountain	Low	Med.	Low	UZ Flow	Unsaturated and Saturated Flow under Isothermal Conditions
	Percolation to depth	Low	Low	Low		
	Seepage into drifts	High	High	High	Seepage	Repository Design and Thermo-mechanical Effects
	Effects of heat and excavation on mountain scale flow	Not Available				
	Effects of heat and excavation on drift scale flow	Low	Med.	Low		
	Dripping onto waste package	Low	Low	Low	Thermal Hydrology Mountain and Drift Scales	Thermal Effects on Flow
	Humidity and temperature at waste package	Low	Low	Low		
Long waste package lifetime	Chemistry of water on waste package	High	Low	Low	Near-Field Geochemical Environment	Evolution of Near-Field Environment
	Integrity of outer carbon-steel barrier	Not Available			Waste Package Degradation	Container Life and Source Term
	Integrity of inner corrosion-resistant barrier	High	High	Med.		
Low rate of release of radionuclides from breached waste packages	Seepage into waste package	Low	Low	Low		
	Integrity of cladding	High	Med.	Med.	Waste Form Degradation, Radionuclide Mobilization, and EBS Transport	
	Dissolution of UO_2 and glass waste form	Low	Med.	Low		
	Solubility of ^{237}Np	Low	Med.	Low		
	Formation and transport of radionuclide-bearing colloids	Low	Med.	Low		
	Transport through and out of the EBS (including waste packages)	Low	Low	Low		
Radionuclide concentration reduction during transport from the waste packages	Transport through the UZ	Low	Low	Low	UZ Transport	Unsaturated and Saturated Flow under Isothermal Conditions and Radionuclide Transport
	Flow and transport in the SZ	Med.	Med.	Med.	SZ Flow and Transport	
	Dilution from pumping	High	High	High		
	Biosphere transport	Med.	Med.	Med.	Biosphere Transport and Uptake	

Source: DOE 1998 [100550], Volume 3, pp. 2-5, 6-12

1.5.5 Summary and Conclusions

This synopsis of individual TSPAs from 1991, 1993, 1995, and 1998, and the clear continuity in how each builds on its predecessors, demonstrates that the general approach and methodology for the TSPA are well established. The DOE has been developing TSPAs for the potential Yucca Mountain site for almost 20 years. While the overall approach has remained the same, the implementation of additional site information, the incorporation of the design into the analyses, and the process models have evolved significantly. Figure 1.5-6 illustrates how the abstractions of components of the system have evolved over the course of the four TSPAs described here. This figure is a composite of Figures 1.5-2, 1.5-3, 1.5-4, and 1.5-5, each of which provides details on progress or remaining issues in each component.

1.6 GENERAL APPROACH FOR CONDUCTING A PERFORMANCE ASSESSMENT

In general, the goal of performance assessment is to provide decision makers with a reasonable estimate of the realistic future performance of the disposal system and a clear display of the extent to which uncertainty in the present understanding of the system affects that estimate. Internationally, most radioactive waste management programs have adopted performance assessments that rely, in one form or another, on computer models as a key element of their safety cases.

Total system performance assessment (TSPA), such as that conducted for the potential Yucca Mountain repository, links models of the components of a disposal system into a single analysis that provides an estimate of overall system performance. Examples of possible system level performance measures that have been adopted or proposed for other repository programs include peak dose to humans from all pathways, cumulative releases of radionuclides from the system, and concentrations of radionuclides in groundwater. For Yucca Mountain, the primary system level performance measures are (1) expected annual dose to humans during the next 10,000 years, and (2) peak concentrations of radionuclides in groundwater (defined in the regulations described in Section 1.3).

Because regulatory requirements specify a consideration of the uncertainty in that estimate, the Yucca Mountain TSPA uses a probabilistic approach similar to that adopted by many other repository programs internationally. This approach has five major steps, shown schematically in Figure 1.6-1 and briefly summarized below. This probabilistic approach was adopted by the EPA in 1985 in the original radiation protection standards, 40 CFR Part 191 (50 FR 38066 [100495]). That rule required a probabilistic performance assessment in the Containment Requirements, with results displayed as a complementary cumulative distribution function showing probability of exceeding specified release limits. Although the currently proposed EPA standards for Yucca Mountain do not include containment limits, a probabilistic performance assessment is still required by the EPA.

1.6.1 Identifying and Screening Potentially Relevant Features, Events, and Processes to Develop Scenarios

The first step in the TSPA is to decide what representations of possible future states of the potential repository (i.e., scenario classes and scenarios) are sufficiently important to warrant quantitative analysis. A further definition of scenario classes and scenarios is found in Appendix A (Glossary). TSPAs can analyze only a relatively small number of the essentially infinite combinations of processes and events that could conceivably affect the system, and it is therefore important that the scenarios chosen for analysis provide a sound basis for evaluating the performance of the site. Specifically, the chosen scenarios must be representative of the conditions of greatest relevance to regulatory requirements and the long-term safety of the site. For the probabilisitic approach required for the Yucca Mountain, estimates must be provided of the probability that the chosen scenarios will occur. Section 2.1 documents the scenario development process adopted for the TSPA-SR, including the basis for identification and screening of potentially relevant FEPs and the selection of the nominal (i.e., undisturbed) and disruptive scenarios classes and their underlying scenarios.

1.6.2 Developing Models

In this step, models are developed to represent components of the system that are potentially important in the chosen scenario classes and scenarios. These models are first developed as conceptual models that describe the behavior of the system. Mathematical models are developed that quantify the conceptual models, and then, in most cases, the mathematical models are implemented in computer codes. (Major models in the Yucca Mountain TSPA are described in Section 3.)

1.6.3 Estimating Parameter Ranges and Uncertainties

Many parameters used in the TSPA models, such as those that describe common rock properties (i.e., porosity and permeability), have natural variability. Uncertainty regarding parameter values also arises from incomplete knowledge (e.g., when the future state of a property must be estimated from assumptions rather than from measurements). In the context of the component models of the TSPA, described in Section 3, uncertainty associated with the selection of parameter values is accounted for by developing distributions of values for important, and imprecisely known, parameters rather than using single values. Each distribution describes a range of values within which the true value is believed to fall, with an expected value (i.e., the mean) that corresponds to the best estimate of the true value. Not all parameters in the TSPA require uncertainty distributions. Single values are used to describe properties that are well known or for which uncertainty has been shown to have little or no effect on overall performance. In cases where realistic uncertainty distributions or parameter values cannot be adequately justified based on available information, parameter distributions or values may be chosen that are deliberately conservative, in the sense that they result in a calculation of performance that is poorer than would result form more realistic input values. The use of conservative or bounding values for input parameters has a potential to mask effects of processes that are treated more realistically, and conservatism should be used cautiously in TSPA.

1.6.4 Performing Calculations

As described in Section 2.2, computer models are linked to allow calculation of the overall system behavior. Uncertainty associated with the selection of scenarios is included in the TSPA by conducting separate analyses for each scenario class. Uncertainty associated with the model parameters is included in the TSPA by conducting multiple calculations for each scenario using values sampled from the ranges of possible values. Each individual calculation uses a different set of sampled input values. In a statistical sense, the result of each individual TSPA calculation represents a different possible realization of the future performance of the system, consistent with the uncertainty in the input parameters.

The expected (i.e., mean) behavior of the system for each scenario class is shown by the mean of the results of the individual calculations, and the uncertainty associated with that mean is shown by the range of calculated outcomes. The overall expected annual dose estimate is determined by summing the expected annual dose calculated for each scenario class, weighted by the probability that the underlying scenarios will occur. The TSPA is, therefore, a probabilistic analysis, consistent with the regulatory requirements described in Section 1.3, in the sense that it takes into account both the estimates of the probability of occurrence of the different scenario classes and scenarios and the uncertainty associated with input parameters.

In addition to the overall TSPA expected annual dose estimate, mean groundwater concentrations are calculated for the nominal scenario to evaluate the groundwater protection performance measure, and a human intrusion expected annual dose is calculated.

1.6.5 Interpreting Results

Results of preliminary performance assessments can be analyzed at the system and subsystem levels to identify the models and parameters that have the greatest effect on the behavior of the system. Identification of the uncertainties that are most important in preliminary TSPAs can help guide testing for site characterization, model development, and repository design. When the TSPA models are sufficiently well developed and documented to support regulatory decisions, results can be used to evaluate compliance with applicable long-term requirements.

1.6.6 Repository Safety Strategy and the Principal Factors

The philosophy and approach to developing the case regarding postclosure safety is presented in the Repository Safety Strategy (CRWMS M&O 2000 [148713]). The postclosure safety case comprises the information the DOE intends to use to assure a potential repository at Yucca Mountain would adequately protect public health and safety after the potential repository is permanently closed. This postclosure safety case is built on a sound technical basis, including information about the Yucca Mountain site, a robust design for the system, comprehensive safety assessments, and assessments of the confidence in that information.

The postclosure safety case described in the Repository Safety Strategy (CRWMS M&O 2000 [148713], Section 1.3) is structured into five elements:

- Performance assessment
- Safety margin and defense-in-depth
- Potentially disruptive processes and events
- Natural analogues
- Performance confirmation.

The safety case focuses on an analysis of 8 factors potentially important to the nominal performance of the potential repository (CRWMS M&O 2000 [148713], Section 4).

1.7 REPOSITORY DESIGN DESCRIPTION

The base case design for the TSPA-SR analyses, the so-called "Reference Design," has recently been updated (CRWMS M&O 2000 [149638]). The basis for the base case design is presented in this section. The design has been formulated with the intention of enhancing system performance with respect to the following key system attributes: 1) water contacting waste package, 2) waste package lifetime (containment), 3) radionuclide mobilization and release, 4) radionuclide transport, and 5) disruptive events and processes. The design strategy seeks to use engineered components to tailor the environmental variables (i.e., temperature, relative humidity, seepage flux) to be as benign as possible.

1.7.1 Base Case Design

A schematic of the base case design at the time of potential repository closure is presented in Figure 1.7-1. In general, the major components of the base case design will include a low areal mass load (60 metric tons of uranium [MTU]/acre), with "line loading" of the waste packages. The EBS will include a drift liner (steel sets with welded wire and rock bolts), an initial air gap in the drift (no backfill), a drip shield, a two-layer waste package (2 cm corrosion-resistant outer material surrounding 5 cm inner structural material), in-drift emplacement of the waste packages, placement of the waste packages on a corrosion resistant emplacement pallet (Alloy-22 and stainless steel), and an invert (steel structure and granular ballast fill [e.g., crushed, welded tuff]) at the base of the drift. The following discussion provides more detail as to the basis for each of these design components.

Drift Support (steel sets with welded wire and rock bolts)–A drift support system has been included in the design, primarily in support of preclosure safety. While the support is intact, it is a potential barrier to seeping water. Seeping water has the potential to drain through the small space between the support and the host rock and also to film flow along the inside surface of the support. Both of these modes of flow reduce or eliminate dripping directly onto the waste package. However, water may also seep through the drift support onto the drip shield.

Air Gap (capillary barrier)–The air gap between the drift support or drift wall and the waste packages provides a means by which percolation flux may be diverted around the opening as matrix or film flow. This advantageous property will be in effect until the drift collapses and fills with rubble.

Drip Shield (titanium alloy)–The drip shield is continuous in the drift over the waste packages. It serves to reduce the effect of rock fall and dripping on the waste package. There is also an initial air gap between the drip shield and the waste package. This will be in effect until the drip shields degrade or collapse onto the waste packages.

Waste Package Corrosion Resistant Material–The outer layer of the waste package is a nickel-based alloy that is very resistant to aqueous corrosion and nearly totally resistant to humid air corrosion. The current reference corrosion-resistant material is 2 cm of Alloy-22.

Waste Package Structural Material–Because the waste package is the single component that is expected to have absolute containment at the time of emplacement, the design strategy is to make the waste package robust. The inner structural material is 5-cm thick and serves three functions.

First, it provides structural strength to resist rock falls, to support the internal components, to be supported by the emplacement pallet and to be handled. Second, the inner layer provides radiation shielding to reduce the waste package exterior surface contact dose rate. Coupled with the Mined Geologic Repository shielded transport, the shielding is enough to protect workers. Third, the inner layer acts as a limited containment barrier for the radioactive waste inside the waste package. The current structural material in the design is stainless steel.

Large Waste Packages–A large waste package reduces cost, handling, closure operations, nondestructive evaluation operations, and allows efficient use of the drift length. The current large waste package reference design is based on a 21 pressurized water reactor spent nuclear fuel assembly waste package. Roughly the same size waste package can also accommodate 44 of the smaller boiling water reactor (BWR) SNF assemblies, five defense high level waste glass "logs" surrounding a central canister of DOE-owned SNF, or a naval spent nuclear fuel canister. For high-heat-producing or high-criticality-potential assemblies, a smaller waste package, for 12 pressurized water reactor (PWR) SNF assemblies, is used. The number of 12 PWRs is small relative to the overall number of waste packages, so they are accounted for indirectly in the mountain scale model, but not in the drift scale model.

In-Drift Emplacement of Waste Packages–The design calls for in-drift emplacement. This is a consequence of large waste packages being well suited to in-drift emplacement; consequently, the amount of excavation is minimized. Nondrift emplacement would require additional excavation.

Invert–The invert is designed to provide support for the waste package emplacement pallets and the drip shields. It will be composed of granular ballast (e.g., crushed, welded tuff), between steel beams that support the rails used during the preclosure period for waste emplacement and performance confirmation equipment.

Alloy 22 Emplacement Pallet–The emplacement pallets provide support for the waste packages during the preclosure period. The pallets will be emplaced along with the waste packages. Each waste package will rest on one pallet.

Thermal Design-Areal Mass Load (Medium) and Thermal Design-Waste Package Spacing (line)–In this reference case, the capacity of the potential repository is designed for the emplacement of 70,000 metric tons of heavy metal (MTHM) (63,000 MTHM commercial spent nuclear fuel (CSNF) and 7,000 MTHM DSNF and HLW).

An areal mass loading of approximately 60 MTU/acre, combined with preclosure ventilation for at least 50 years from the start of emplacement, will prevent the boiling zones from coalescing in the pillars between emplacement drifts. Waste packages are placed in the emplacement drifts in a line load configuration with a waste package to waste package spacing of approximately 10 cm. The diameter of a waste emplacement drift is 5.5 m. Emplacement drifts are arranged with a uniform spacing of 81 m between their centerlines. The total emplacement drift length is calculated from adding the waste package inventory (including DOE waste packages) and the package-to-package gaps. The emplacement area encompasses 1,050 acres in the upper emplacement level of the characterized area.

Thermal Design-Spent Nuclear Fuel Assembly Blending-To Meet 11.8 kW Limit–Each spent nuclear fuel assembly has a specific set of characteristics: enrichment, burnup, and age. These characteristics determine how much thermal power the assembly produces and the rate of decline of that power. Waste package heat output at emplacement is not to exceed 11.8 kW. This specification requires blending commercial fuel assemblies to no more than 20 percent above the average PWR thermal heat output (9.8 kW per package). The average age of the CSNF is assumed to be 26 years, with no additional aging beyond that imposed by reactor and potential repository operation schedules.

1.7.2 Design Option Sensitivity Cases

The SR base case design does not include backfill in the emplacement drifts. However, one design option that will be analyzed is the case that includes backfill. It provides additional protection of the drip shield from the rock fall. Another design option is a low thermal load option. See Section 4.6 for these analyses.

Backfill Rock Fall Protection–During the postclosure period, the drift ground support and parts of the near-field rock may fall into the drift. These rock falls have the potential to affect the performance of the drip shield. Analyses have concluded that the probability of generating a through-going crack on a drip shield is negligible (CRWMS M&O 2000 [149574]). Even if a crack is generated, the in-growth of corrosion products and calcite deposition are expected to effectively seal the crack (CRWMS M&O 2000 [151949]). Therefore the effects of rock fall on system performance have been screened out of the TSPA analyses.

1.8 SITE DESCRIPTION

Characteristics of the natural system at Yucca Mountain, such as its semiarid climate and deep water table, led, in part, to its current consideration as the setting for a potential geologic repository for HLW. These characteristics provide an environment that could potentially isolate the waste from the effects of water for long periods.

1.8.1 General

Characteristics of the natural system at Yucca Mountain that may affect potential repository performance include climate, site geology, and site hydrogeology. Characteristics of the site geology and hydrogeology that will affect potential repository performance include groundwater flow through the UZ and SZ, radionuclide transport, and disruptive events such as volcanism and earthquakes. Other factors considered and analyzed when describing the site are population distribution and land use around Yucca Mountain. Figure 1.8-1 provides an illustration of the processes affecting potential repository performance, including physical characteristics of Yucca Mountain. This section provides a brief description of the current understanding of the natural system at Yucca Mountain and is based on Section 2 of the Viability Assessment (DOE 1998 [100548]). Detailed information on Yucca Mountain site characteristics and descriptions of the investigations conducted at Yucca Mountain can be found in the *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]) that comprehensively reports current knowledge about the site and its surrounding region.

Both surface-based and underground investigations have been used to characterize the Yucca Mountain site. These studies have included the following:

- Monitoring the present meteorology to support development of infiltration models, completion of environmental analyses, design of potential repository facilities, and calculation of atmospheric dispersion
- Mapping geologic structures, including rock units, faults, fractures, and volcanic features
- Using gravitational, magnetic, electrical, and seismic methods to infer the distribution and properties of geologic units and structures at depth
- Monitoring earthquake activity
- Characterizing earthquake faults at the potential repository horizon
- Drilling boreholes into the mountain to identify the geologic units present, measure the depth to groundwater, measure the properties of the hydrologic system, and determine air- and water-movement properties above the water table
- Heating a large block of Topopah Springs tuff to observe the subsequent effects of heat on its hydrologic and chemical properties
- Excavating a 7.9-km main tunnel Exploratory Studies Facility Main Loop) into Yucca Mountain adjacent to the potential repository site and second tunnel 2.7-km long crossing 15-20 m above the potential repository area in a southwest direction
- Monitoring seepage into the Exploratory Studies Facility, including the effects of ventilation and injected pulses of water
- Conducting two thermal tests in the Exploratory Studies Facility at the potential repository horizon.

1.8.2 Physiography

Yucca Mountain is located in southern Nevada approximately 161 km northwest of Las Vegas (Figure 1.8-2). The mountain is an irregularly shaped volcanic upland varying in elevation at its crest from 1,500 m to 1,930 m above mean sea level and characterized by approximately 650 m of relief. An aerial photograph of Yucca Mountain is provided as Figure 1.8-3, showing the location of the entrance to the Exploratory Studies Facility Main Loop.

Yucca Mountain is located in the Basin and Range province of the western United States, within the region known as the Great Basin (Figure 1.8-4). The Great Basin encompasses nearly all of Nevada and parts of Utah, Idaho, Oregon, and California. The mountain ranges of the Great Basin are mostly north-south aligned, tilted, fault-bounded blocks that may extend more than 80 km in length and are generally 8 to 24 km wide. Relief between valley floors and mountain ridges is typically 300 to 1,500 m, and valleys occupy approximately 50 to 60 percent of the total land area. The valleys are filled with thick deposits of alluvium derived from erosion of the adjacent ranges. In general, the ranges are separated (north and south) by roughly 25 to 30 km. However, some ranges arc toward each other and merge together.

1.8.3 Land Use and Population Density

Yucca Mountain occupies land controlled by three federal agencies: the U.S. Air Force (i.e., Nellis Air Force Range), DOE (i.e., Nevada Test Site), and the U.S. Bureau of Land Management. Nearly all the area surrounding Yucca Mountain is federally owned, and very little is developed or urban land. A large percentage of the land around Yucca Mountain is anticipated to remain federally owned or withheld from public use in the future. The area surrounding the site includes Nye, Lincoln, Esmeralda, and Clark Counties in Nevada and Inyo County in California (Figure 1.8-4).

Population density near Yucca Mountain is low. Nye County, which encompasses the site, has 0.62 persons per square kilometer. Of the total population of 29,730 in Nye County, 68 percent live in the unincorporated town of Pahrump, 70 to 80 km south-southeast of Yucca Mountain (CRWMS M&O 2000 [137917], p. 2.3-1). Larger concentrations of population are found in Clark County to the southeast, in the incorporated and unincorporated areas of the Las Vegas valley.

1.8.4 Climate

Climate, and its changes over time, directly affects system performance at Yucca Mountain.

Precipitation and surface weather conditions limit the infiltration of water into and through the mountain. While the Yucca Mountain climate is currently very dry and hot, past climate records show this was not always the case. Future climate will likely be similar to past climates, which have been wetter and cooler than that of the present.

1.8.4.1 Present-Day Climate

Present-day climate in southern Nevada is semiarid, with hot summers and mild winters. Local and regional monitoring stations provide weather data for the Yucca Mountain vicinity. Average annual precipitation over a 30-year period for meteorological stations in the Yucca Mountain area range from 112 mm to 144 mm (CRWMS M&O 2000 [137917], Table 6.2-3). The estimated annual potential evapotranspiration in Las Vegas is 1,067 mm per year (Houghton et al. 1975 [106182], p. 63, Figure 61). Snowfall is infrequent, light, and short lived below about 1,070 m above mean sea level. The estimated maximum daily rainfall is bounded by a value of 125 mm (CRWMS M&O 1997 [100117], pp. 4 to 21).

The annual average temperature in the Yucca Mountain area ranges from about 15° to 18°C, depending on elevation (CRWMS M&O 2000 [137917], Section 6.2.3.2). Summer temperatures can exceed 40°C and winter temperatures occasionally fall below 0°C. The annual average dew point temperature is about -5°C. Regional weather systems and the mountain and valley topography cause a regular wind pattern of well-mixed airflow toward the north during the daytime and stable, low-mixing airflow toward the south into Amargosa Valley at night.

Aridity and warm temperatures result from a combination of large-scale atmospheric circulation patterns and the large mountain ranges, such as the Sierra Nevada, on the pathway from the primary moisture source, the Pacific Ocean. The Yucca Mountain area is affected by typical midlatitude global circulation patterns, with weather systems moving from west to east. Storms moving into the area from the southwest during winter tend to have the greatest potential for high precipitation as rain or snow. Significant late summer, southwest monsoon precipitation events occur with moist airflow from the south, originating either in the Gulf of Mexico or the Pacific Ocean. Naturally recurring short-term changes in typical circulation patterns alter storm paths and precipitation patterns. One example is the El Niño pattern, which tends to increase winter precipitation in Southern Nevada by approximately 50 percent.

1.8.4.2 Paleoclimate

Evidence of the cyclic nature of climate comes from long-term climate records. Ongoing scientific studies indicate that microfossils and stable isotopes from oceanic and lacustrine sediments vary in response to climate change and act as proxies or substitutes for direct observation of past climates.

Long-Term Climatic Records–Three long-term climatic records occur within 100 miles of Yucca Mountain: Devils Hole, Owens Lake, and Death Valley.

Devils Hole (about 60 km southeast of Yucca Mountain) yields a well-dated, stable isotope record from calcite that was deposited on the submerged walls of a fissure within the regional carbonate aquifer (Winograd et al. 1996 [109468]). The Devils Hole record, with its extensive and accurate chronology, establishes the timing and duration of glacial periods in the Yucca Mountain region (CRWMS M&O 2000 [137917], p. 6.3-11).

Owens Lake contains a long record of lacustrine sediments containing fossil and stable isotope evidence of climate change. Owens Lake is a present-day playa in Inyo County, California, approximately 160 km west of Yucca Mountain. Interpretation of the Owens Lake climate record provides information about the magnitude of change in precipitation and air temperature during past climate periods and, therefore, complements the Devils Hole record (Smith and Bischoff 1997 [100077]). Over the past 400,000 years, Owens Lake has been fresh (implying a glacial climate, wetter than present) for approximately 80 percent of the time and saline (implying climates like the present) 20 percent of the time (CRWMS M&O 2000 [137917], p. 6.3-17).

Data from a 185-m sedimentary core taken in Death Valley, California, spanning a time period of about 200,000 years, indicate that this area was filled by a large, deep lake during an earlier glacial period (Li et al. 1996 [100054]). The persistence of lakes during cooler and wetter climate periods illustrates that effective moisture levels were much higher during glacial periods.

Short-Term Climatic Records–A number of important short-term climatic records exist within the Yucca Mountain area. They include pack-rat middens, paleowetland deposits, and paleospring deposits. These records document past climates in the Yucca Mountain region during the last 50,000 years before the present.

Pack-rat middens are deposits of fossil plant remains and other material cemented by crystallized urine. Analysis of the middens provides information on climate conditions over time, because the plants available to the rats are indicative of existing climate conditions and because the plants and other organic matter can be dated by radiocarbon techniques. Studies of plant fossils from middens in the Yucca Mountain area from approximately 35,000 to 12,000 years ago reveal that conditions during the last glacial period were cooler and wetter than today (Spaulding 1990 [100078], Chapter 9).

Sedimentary deposits found on valley floors throughout southern Nevada indicate that during the last glacial period there were wetlands, flowing springs, and streams at low elevation (Quade et al. 1995 [100074]). Aquatic vegetation was common on the alluvial fan deposits sloping down from adjacent mountains and in the wetlands. The springs and wetland sediments contain vertebrate fossils that indicate cooler, wetter conditions existed. The existence of wetlands throughout the region, together with the types of fossils found, indicates that recent glacial periods were colder and wetter than today.

Interpretation of data from the Crater Flat Deposit, a paleospring deposit located approximately 15 km southwest of Yucca Mountain, suggests the regional water table was at the surface at this site during glacial periods (Paces, Forester et al. 1996 [101281], Section 2.2). Today, the regional water table at this location is approximately 100 m below ground surface.

Site Records of Climate Change–Stable carbon and oxygen isotope values of calcite that precipitated in fractures within Yucca Mountain provide potential climate information related to infiltration. Quade and Cerling (1990 [100073], pp. 1549 to 1552) concluded that the carbonate in pedogenic calcrete filling the Bow Ridge fault formed during climates that were colder and wetter than the present. The authors correlated these carbon isotope values to those of modern soil carbonate forming at elevations of 1,800 to 2,000 m, which is comparable in today's climate to the flanks of Rainier Mesa.

1.8.4.3 Future Climate

Forecasting future climate depends on the assumption that climate is related to measurable and predictable processes, such as the variation of insolation caused by changes in earth orbit. Assumptions used for forecasting future climate include (CRWMS M&O 2000 [137917], Section 6.4.3):

- Climate is cyclical, and past climates provide insight into potential future climates.
- A relationship exists between the timing of past long-term climate change and earth-orbital cycles.
- A relationship exists between the characteristics of past climates and the sequence of those climates in the 400,000-year long-term earth-orbital cycle.
- Long-term, Earth-based, climate-forcing processes, such as tectonics, have remained relatively constant over the past 400,000 years or so and should remain so for the period of interest for performance assessment.

Based on these assumptions and the long-term paleoclimate records described above, the following forecast for future climate at Yucca Mountain has been developed: the modern-day climate at Yucca Mountain should persist for 400 to 600 years, followed by a warmer and much wetter monsoon climate for 900 to 1,400 years, followed by a cooler and wetter glacial-transition climate for 8,000 to 8,700 years (CRWMS M&O 2000 [137917], Section 6.5.3).

1.8.5 Geology

The understanding of Yucca Mountain geology has evolved following years of extensive studies, resulting in the construction of a detailed, integrated site geological model (CRWMS M&O 2000 [146988]) containing stratigraphic and structural relationships, as well as rock-property and mineralogical data.

1.8.5.1 Regional Geology

Yucca Mountain is in the Great Basin Region of the Basin and Range Province, which is characterized by mostly north-south aligned, tilted, fault-bounded blocks, with wide valleys filled with thick deposits of alluvium derived from erosion of the adjacent ranges. This pattern is the result of generally east-west-directed crustal extension that began in the Tertiary period and continues at present (Hamilton 1988 [100037], Chapter 5). The extension has caused complex faulting, resulting in ranges composed of tilted fault blocks bounded by major range-front faults. Seismic reflection geophysical studies show this style of deformation extends beneath the intervening basins, where it is buried by alluvium (Brocher et al. 1998 [100022], pp. 947 to 971). Rocks from Precambrian to Quaternary in age have been deformed by this extension.

Yucca Mountain lies within the Walker Lane structural domain, which extends northwestward from Las Vegas, parallel to the Nevada-California border. This structural domain is characterized by northwest-trending, right-lateral faults and northeast-trending, left-lateral faults (Stewart 1988, [100083] p. 3).

The Inyo-Mono domain to the southwest of Yucca Mountain has the highest rate of seismic and volcanic activity in the southwestern Great Basin and is an important part of the regional geologic setting. This domain includes Death Valley Basin and the Panamint Range (CRWMS M&O 2000 [137917], Section 4.2.1.3).

Due to the deep-seated block faulting, rocks from the Precambrian Era through recent sedimentary and volcanic deposits are exposed in the Great Basin:

- Precambrian rocks (540 Ma to 4550 Ma) include an older, metamorphosed assemblage and a younger, metasedimentary assemblage (CRWMS M&O 2000 [137917], Section 4.2.2.1.1). Both groups tend to retard the groundwater flow where extensive faulting or fracturing is present.

- Paleozoic rocks (250 Ma to 540 Ma) include older carbonate strata; middle, fine-grained shale, siltstone, and sandstone unit; and an upper carbonate unit. The carbonate units form important aquifers throughout southern Nevada (Winograd and Thordarson 1975 [101167], pp. C9 to C12).

- Mesozoic rocks (65 Ma to 250 Ma) are generally absent near Yucca Mountain (CRWMS M&O 2000 [137917], Section 4.2.2.1.3) because the Mesozoic was a period of active tectonic activity characterized by regional compression in this area.

- Cenozoic rocks (Present to 65 Ma) near Yucca Mountain fall into three groups: premiddle Miocene sedimentary rocks, mid-to-late Miocene volcanic (including Yucca Mountain), and Plio-Pleistocene basalt and basin sediments (CRWMS M&O 2000 [137917], Section 4.2.2.2).

1.8.5.2 Site Stratigraphy

The entire sequence at Yucca Mountain is composed of mid-to-late Miocene volcanic rocks formed by eruptions of ash or magma from volcanic vents to the north (Sawyer et al. 1994 [100075], pp. 1304 to 1318; Buesch et al. 1996 [100106]). Most of the rocks are ash flow tuffs, which occur as welded tuffs, nonwelded tuffs, air-fall tuffs, or bedded tuffs (reworked by stream action). Figure 1.8-5 is a simplified cross section through Yucca Mountain and depicts the major units present. Stratigraphic units relevant to the potential repository are the Paintbrush Group, the Calico Hill Formation, and the Crater Flats Group. The Paintbrush Group is subdivided into the Tiva Canyon tuff, Yucca Mountain and Pah Canyon tuffs (the latter two are referred to as the pre-Tiva Canyon tuff in Figure 1.8-5), and Topopah Spring tuff (CRWMS M&O 2000 [137917], Section 4.5.4.7). The potential repository is located within the Topopah Spring tuff.

The Tiva Canyon tuff is a large-volume, regionally extensive ash flow tuff that comprises most of the rocks exposed at the surface of Yucca Mountain. The unit is divided into two members with differing chemical compositions: a lower crystal-poor rhyolite member and an upper crystal-rich quartz-latite member (Buesch et al. 1996 [100106], pp. 16 to 18, Figure 2). The thickness of the formation ranges from less than 50 m to as much as 175 m respectively. The Yucca Mountain and Pah Canyon tuffs vary in texture from nonwelded to densely welded and in thickness from 0 to 77 m (Buesch et al. 1996 [100106], pp. 18 to 19).

The Topopah Spring tuff, which is the host rock for the potential repository, has a maximum thickness of about 380 m near Yucca Mountain (Buesch et al. 1996 [100106], pp. 19 to 21). Like the Tiva Canyon tuff, the Topopah Spring tuff is compositionally zoned from a lower crystal-poor (less than 10 percent phenocrysts) rhyolite to an upper crystal-rich quartz latite. Each member is further divided by degree of vitrification and the abundance of lithophysae (voids in the rock caused by bubbles of volcanic gases trapped in the rock matrix during cooling). The crystal-poor member is divided into a vitric zone near the base and devitrified rocks of the lower nonlithophysal, lower lithophysal, middle nonlithophysal, and upper lithophysal zones. The latter three zones are densely welded and comprise the potential repository horizon. The nature, size, and abundance of lithophysae in the tuffs are important because they may affect the mechanical and hydrologic properties of the rock.

The Calico Hills Formation is a series of rhyolite tuffs and lavas (Sawyer et al. 1994 [100075], p. 1307) and is significant to the potential repository because of its hydraulic properties and mineralogical composition. The formation thins southward, from a total thickness of 460 m north of the potential repository block, to 15 m to the south (CRWMS M&O 2000 [137917], Section 4.5.4.6). None of the Calico Hills ash flows are strongly welded; therefore, the rocks have much lower strength and higher porosity than the Topopah Spring tuff. Because of their lower strength, fractures are rare or absent in the Calico Hills. The sparsity of continuous fracture pathways may have important implications for water flow in the UZ. Another important attribute of the Calico Hills is an abundance of zeolite minerals (Broxton et al. 1993 [107386], pp. 19 to 22). Zeolites have the ability to sorb radionuclides that may be transported in water. Sorption may significantly slow the movement of many radionuclides away from a potential repository.

The Crater Flat Group consists of three sequences of rhyolitic, moderate- to large-volume ash flow deposits and interlayered, bedded tuffs. In descending order, these formations are the Prow Pass, Bullfrog, and Tram tuffs (Sawyer et al. 1994 [100075], Table 1). The Prow Pass tuff is a sequence of variably welded ash flow deposits ranging from about 60 to 228 m in thickness and commonly zeolite-altered. The Bullfrog tuff consists of upper and lower zones of welded to partially welded zeolite-altered tuff, separated by a central zone of moderately to densely welded tuff. The measured thickness of the entire sequence ranges from 76 m to as much as 400 m. The Tram tuff includes a lower, lithic-rich unit and an upper, lithic-poor unit. The lithic-poor unit is normally more densely welded than the underlying lithic-rich unit. Both units contain clay and zeolite alteration. The thickness of the Tram tuff ranges from about 60 to 396 m (CRWMS M&O 2000 [137917], Section 4.5.4.5).

1.8.5.3 Site Structural Geology and Earthquake Hazard

Yucca Mountain is part of a down-dropped block bounded on the west by a steeply eastward dipping fault, the Bare Mountain fault. Beneath Crater Flat, the block is segmented by a series of eastward dipping faults, and beneath Yucca Mountain and Jackass Flats generally westward dipping faults are interpreted. The block-bounding faults of Yucca Mountain (e.g., the Solitario Canyon, Bow Ridge, Paintbrush Canyon, and Windy Wash faults) and the Bare Mountain fault appear to be discrete, planar faults, at least some of which may descend to the base of the earth's brittle crust. Locations of the major faults at Yucca Mountain are shown on Figure 1.8-6.

Faults within 100 km of Yucca Mountain have been examined using aerial photographs. All faults with suspected Quaternary movement were evaluated. Natural exposures were cleared, and approximately 60 trenches have been excavated across faults within and near the site. Information from these trench studies indicates that estimated slip rates for faults at Yucca Mountain are low, varying from 0.0001 mm per year to 0.03 mm per year (Whitney et al. 1996 [107313], pp. 5 to 11). The average time interval between surface displacement events varies from 13,000 to 100,000 or more years (Whitney et al. 1996 [107313], pp. 5 to 11). Offsets of the earth's surface range from 3 to 167 cm per large event (Whitney et al. 1996 [107313], Table 5-1).

Vibratory ground motion and fault displacement hazards at Yucca Mountain have been analyzed probabilistically. A preliminary probabilistic study was carried out to support design of the Exploratory Studies Facility (CRWMS M&O 1994 [100112], Table 1). Results indicated ground motions of 265 cm per square second and 647 cm per square second are expected to be exceeded on average every 1,000 and 10,000 years, respectively. This earlier analysis of seismic hazards was updated by two expert panels (USGS 1998 [100354], CRWMS M&O 2000 [142321]). To determine ground motion and fault displacement hazard at Yucca Mountain, the experts' assessments were integrated along with their evaluations of uncertainties. The vibratory ground motion hazard was computed at a reference rock outcrop, which corresponds to the proposed waste emplacement depth. Ground motion was computed at this reference location as a control motion to facilitate the subsequent determination of design basis motions for surface locations and potential waste-emplacement level locations following completion of geotechnical investigations. For peak ground acceleration, results with a 10^{-3} and 10^{-4} annual frequency of being exceeded are, respectively, 165 and 523 cm per square second for the horizontal component and 108 and 383 cm per square second for the vertical component (USGS 1998 [100354], Table 7-1).

Probabilistic analysis based on expert assessments indicates that geologic fault displacement hazard is generally low. For sites not located on a major block-bounding fault, displacements greater than 0.1 cm will be exceeded on average less than once in 100,000 years. For this same time period, the mean displacements that are expected to be exceeded on two of the block-bounding faults (Bow Ridge and Solitario Canyon faults) are 7.8 and 32 cm, respectively (USGS 1998 [100354], Table 8-1). The primary design approach to mitigate fault displacement effects involves avoiding faults in laying out potential repository facilities.

Modern seismicity has been monitored at the Nevada Test Site since 1968. In 1979, a network of seismic stations was established in the southern Great Basin to monitor earthquakes near Yucca Mountain (Rogers et al. 1987 [100176], p. 3). The largest earthquake detected by the monitoring network was the magnitude 5.6 event near Little Skull Mountain, located about 12 mi north of Yucca Mountain, on June 29, 1992.

1.8.5.4 Volcanology and Volcanic Hazard

Rocks composing Yucca Mountain are part of the southwestern Nevada volcanic field. This field was formed by the eruption of large volumes of volcanic rocks from multiple sources to the north. The explosive volcanism that culminated in the formation of the southwestern Nevada volcanic field is the most significant depositional event of the Cenozoic era near Yucca Mountain. This event formed six major calderas between 15 million and 7.5 million years ago (Sawyer et al. 1994 [100075], p. 1,304). This event also created the rocks of Yucca Mountain.

The most recent events are infrequently erupted basaltic volcanic rocks. The basaltic eruptions represent a continuation of the activity during the mid- to late-Miocene epoch (CRWMS M&O 2000 [141044], Section 6). Following an episode 3.7 million years ago, a subsequent basaltic eruption occurred between 1.7 million and 0.7 million years ago consisting of four cinder cones (Little Cone, Red Cone, Black Cone, and Makani Cone) aligned north-northeast along the Crater Flat axis. The final episode of basaltic volcanism created the Lathrop Wells Cone, which includes fissure eruptions, spatter and scoria cones, and basaltic lava flows. The Lathrop Wells Cone complex is approximately 80,000 years old (CRWMS M&O 2000 [137917], p. 4.10-2). The eruption volume of individual basaltic volcanic events has also been decreasing progressively through time. The decreased rate of volcanic activity correlates with the decreased rate of regional extension and faulting.

To assess the likelihood of volcanic activity disrupting a potential repository, a panel of ten experts representing a wide range of expertise in the fields of physical volcanology, volcanic hazards, geophysics, and geochemistry conducted an assessment (CRWMS M&O 1996 [100116]). The scientists reviewed extensive information presented by representatives of DOE, U.S. Geological Survey, the State of Nevada, NRC, and others regarding the spatial and temporal distribution of future volcanic activity near Yucca Mountain. That work was supplemented by analyses conducted for the TSPA-SR (CRWMS M&O 2000 [141044], Section 6). A probability distribution, revised for the current potential repository footprint, gives a mean value of 1.6×10^8, which corresponds to approximately one chance in 6,250 of a volcanic event (dike intrusion) disrupting the potential repository during the first 10,000 years after closure.

1.8.6 Hydrogeology

1.8.6.1 Regional Hydrogeology

Yucca Mountain lies within the Alkali Flat-Furnace Creek groundwater subbasin, one of several that comprise the Death Valley regional flow system (Luckey et al. 1996 [100465], p. 13). The subbasins are structural basins formed during mid-Tertiary crustal extension that are filled with gravel, sand, silt, and clay eroded from the adjacent uplands, forming a major aquifer hundreds of meters thick.

Recharge to the northeastern quadrant of the Death Valley system occurs principally at higher elevations. The area north of Yucca Mountain, including Central Pahute Mesa, Timber Mountain, and Shoshone Mountain provides most of the recharge to the Alkali Flat-Furnace Creek subbasin. Regional water level contours show a southward slope of the water table from recharge areas in the northern part of the Alkali Flat-Furnace Creek subbasin toward discharge areas in the southern Amargosa Desert. The Yucca Mountain site occupies an intermediate position along this path in an area where the contours indicate a probable southeastward flow direction. North-south and northwest-southeast oriented faults and fractures probably assert a strong influence on flow direction and contribute to continued southerly flow.

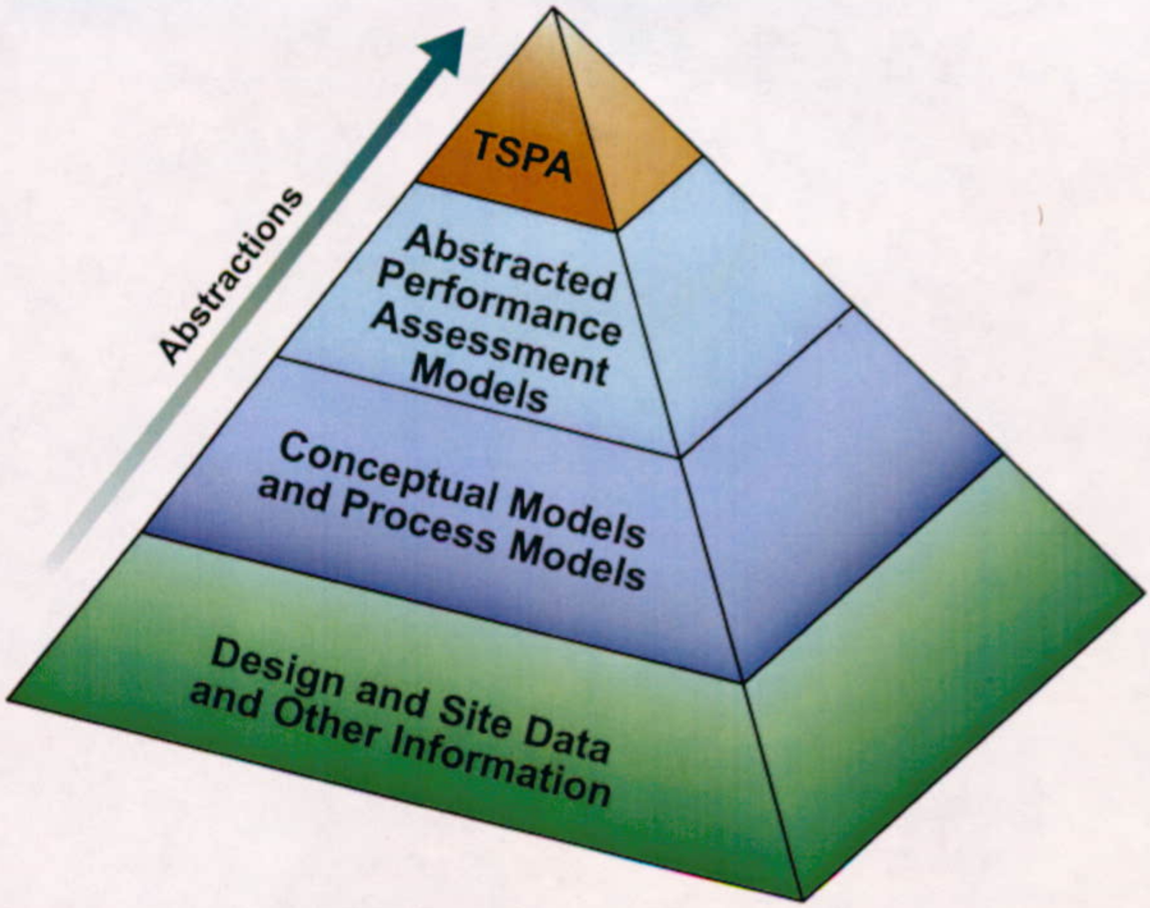
The dominant regional aquifer underlying the southern part of the Alkali Flat-Furnace Creek subbasin is in Paleozoic marine limestones, dolomites, and minor clastic sediments that are thousands of meters thick (carbonate aquifer). Fractures enlarged by dissolution provide the large permeability associated with this aquifer. The carbonate aquifer hydrologically connects many valleys and intervening ranges. Beneath the carbonate aquifer are relatively impermeable, metamorphosed older rocks, known as the lower clastic aquitard or the Paleozoic-Precambrian clastic confining units (D'Agnese et al. 1997 [100131], p. 20). These rocks form the effective hydraulic basement for groundwater flow.

1.8.6.2 Site Hydrogeology

Results of hydrologic investigations at and near Yucca Mountain have been summarized (Luckey et al. 1996 [100465]). Except for one borehole, the drilling program has not reached the base of the Tertiary volcanic section. North of the potential repository, the volcanic rocks are at least 1,829 m thick. Near the southern boundary, drilling has established a minimum depth of 1,533 m. Based on rock type and flow properties, Luckey et al. (1996 [100465], pp. 18 to 20, Figure 7) have divided the volcanic rocks below the water table into four hydrogeologic units. These are known, from the top down, as the upper volcanic aquifer, the upper volcanic confining unit, the lower volcanic aquifer, and the lower volcanic confining unit. The upper volcanic aquifer is composed of the Topopah Spring welded tuff, which occurs in the UZ near the potential repository but is present beneath the water table to the east and south of the potential repository and in Crater Flat. The upper volcanic confining unit includes the Calico Hills nonwelded unit and the uppermost, unfractured part of the Prow Pass tuff where they are saturated. The lower volcanic aquifer includes most of the Crater Flat Group, and the lower volcanic confining unit includes the lowermost Crater Flat Group and deeper tuffs, lavas, and flow breccias.

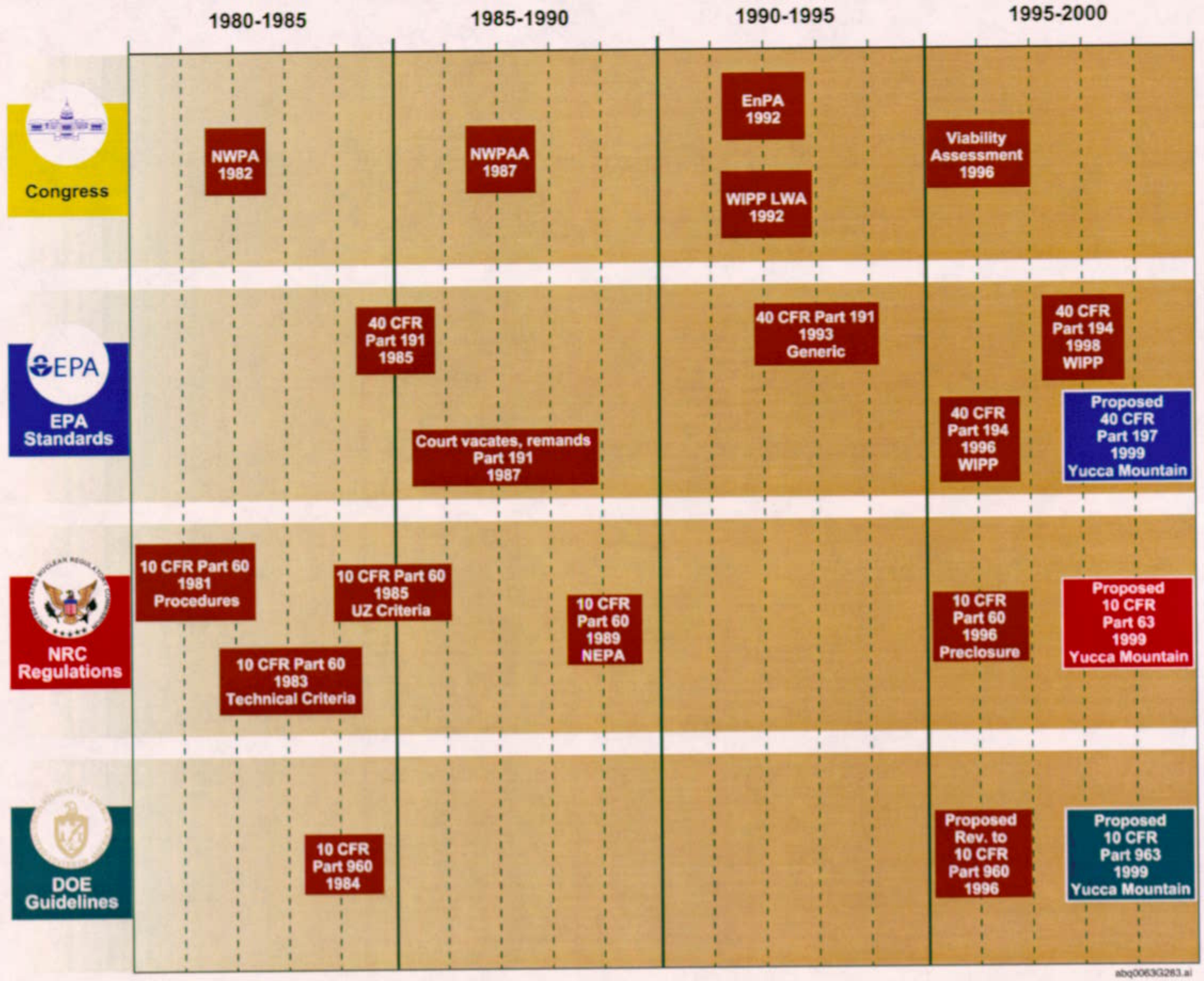
The main distinction between volcanic aquifers and confining units is that the aquifers tend to be more welded and contain more permeable fractures. However, alteration of the tuffs to zeolites and clays, which reduces permeability, is more pronounced at depth, and the greater pressure tends to reduce fracture permeability. Consequently, a combination of factors including fracture properties, mineralogy, and depth, rather than just rock type, determines the hydrologic character of the volcanic rocks below the water table at Yucca Mountain.

Hydraulic tests have been performed to determine the properties of the units. The analyses are limited by significant uncertainties about the extent to which fractures affect the unit hydraulic conductivity (Luckey et al. 1996 [100465], pp. 32 to 36). However, ranges of hydraulic conductivity values are reported to provide comparison among the units. The confining units have low hydraulic conductivity that ranges from 0.0000055 m/day to a maximum of 0.26 m/day respectively, with the aquifers ranging from 0.0037 to 1.4 m/day respectively (Luckey et al. 1996 [100465], Table 4).



abq0063G312

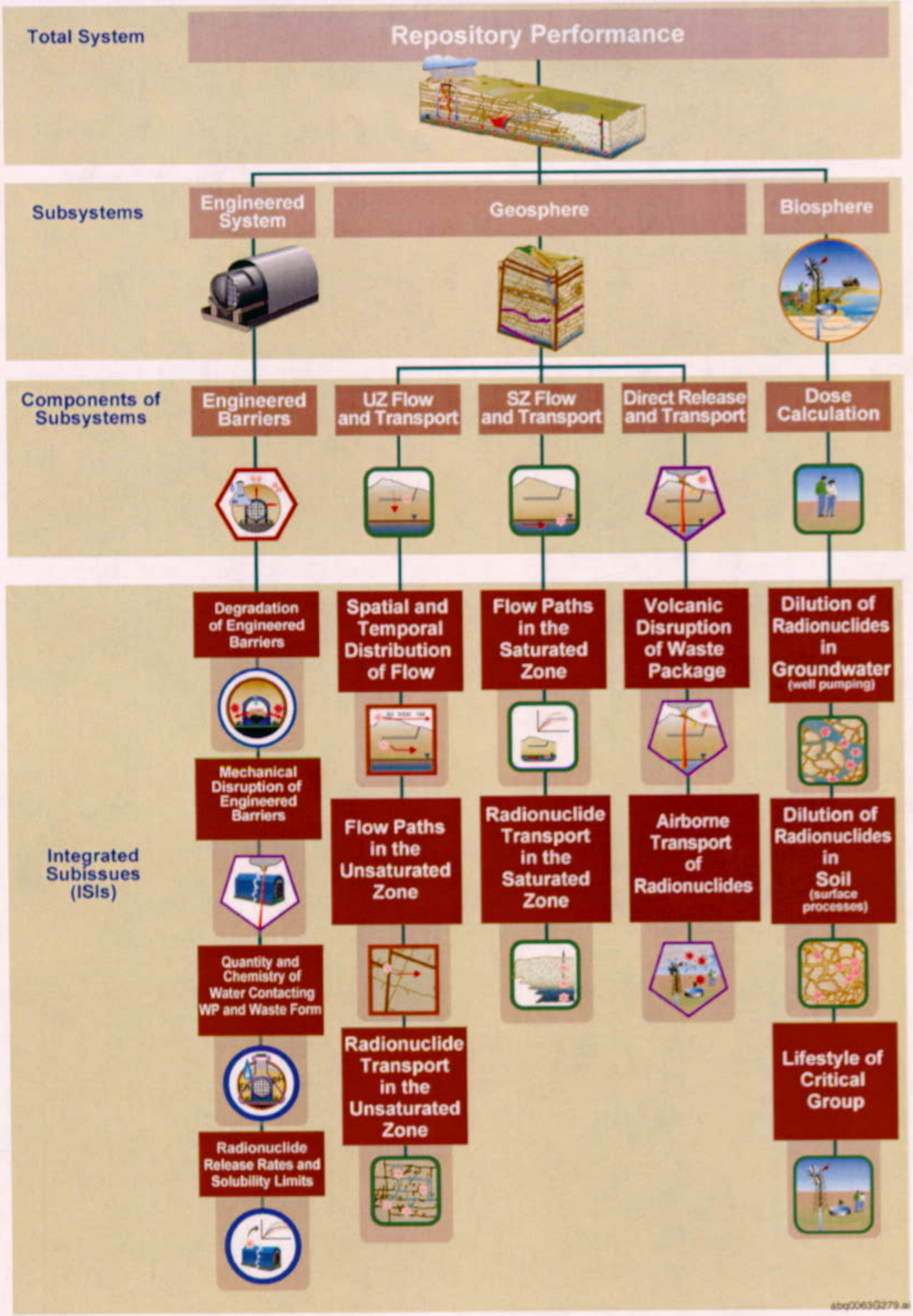
Figure 1.1-1. Total System Performance Assessment Information Pyramid



abq0063G283

NOTE: NWPA – Nuclear Policy Act of 1982, NWPAA – Nuclear Waste Policy Amendment Act of 1987, EnPA – Energy Policy Act of 1992, WIPP LWA – Waste Isolation Pilot Plant Land Withdrawal Act of 1992.

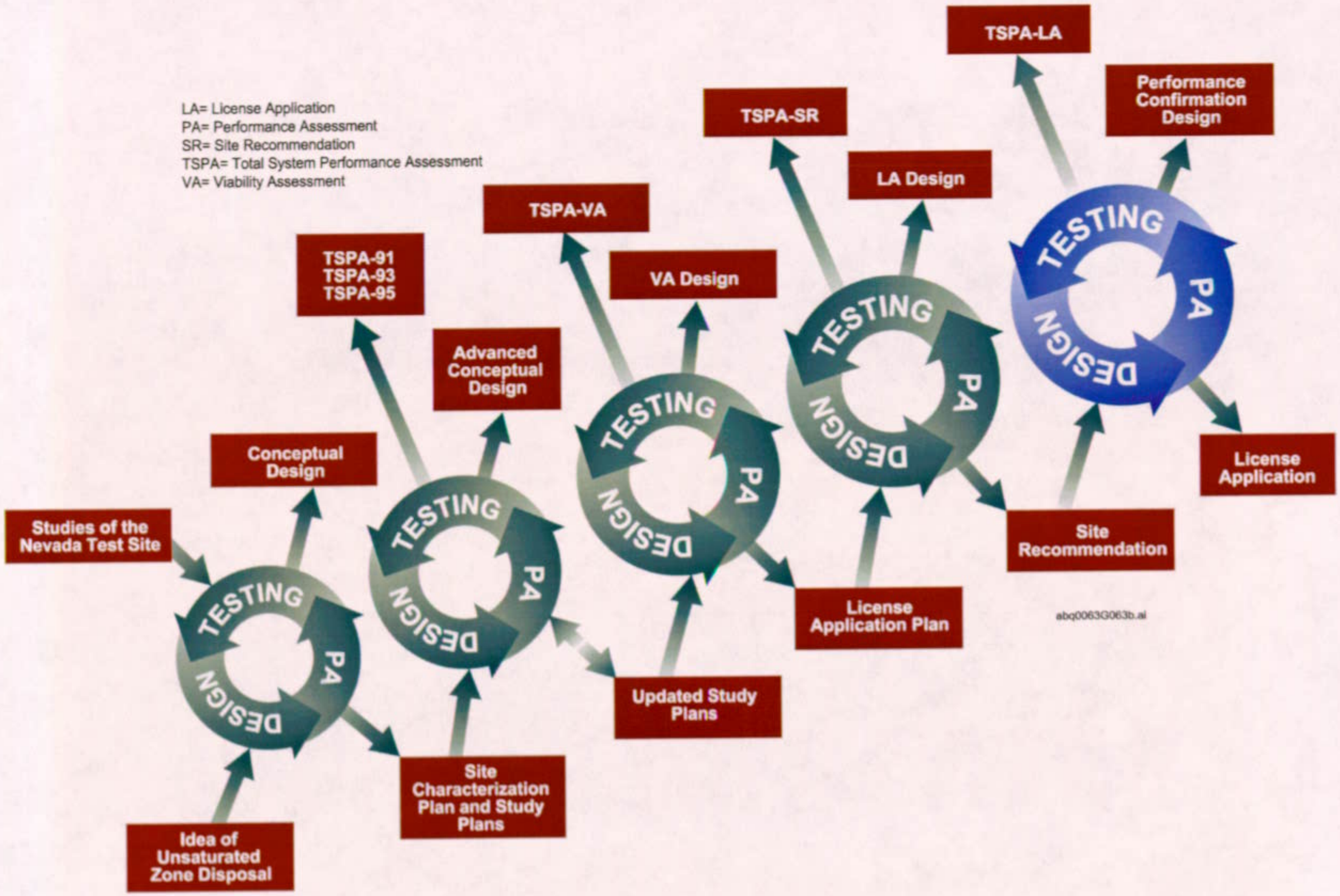
Figure 1.3-1. Timeline of Legislative and Regulator Events, 1980 to 2000



abq0063G279

Source: NRC 2000 [149372]

Figure 1.3-2. Integrated Subissues of Model Abstraction Subissue for Total System Performance Assessment and Integration Issue Resolution Status Report



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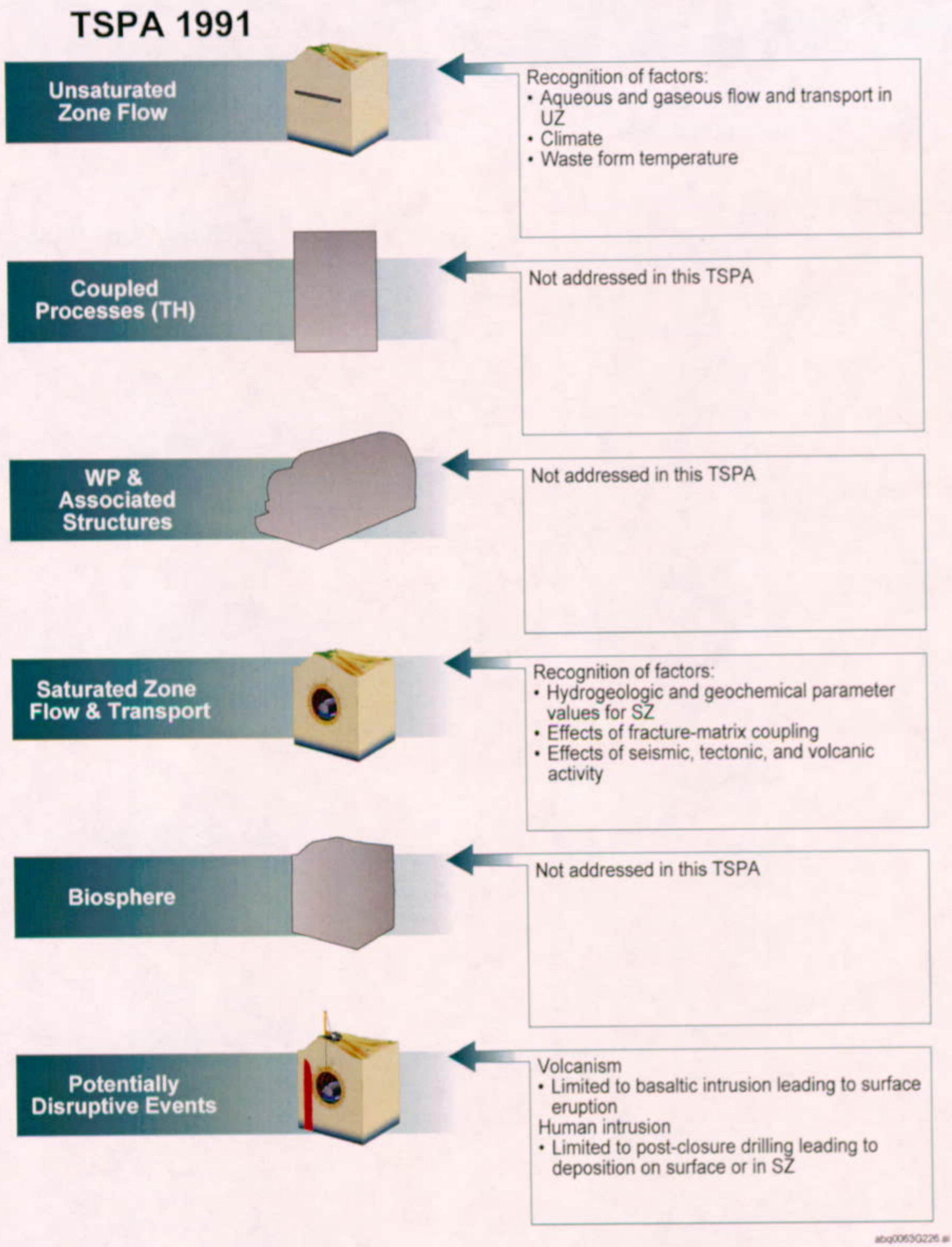
Source: DOE 1998 [100550], Volume 1, Figure 1-4, p. 1-21.

Figure 1.5-1. Iterative Application of the Total System Performance Assessment Tool to Advance Understanding of the Yucca Mountain System



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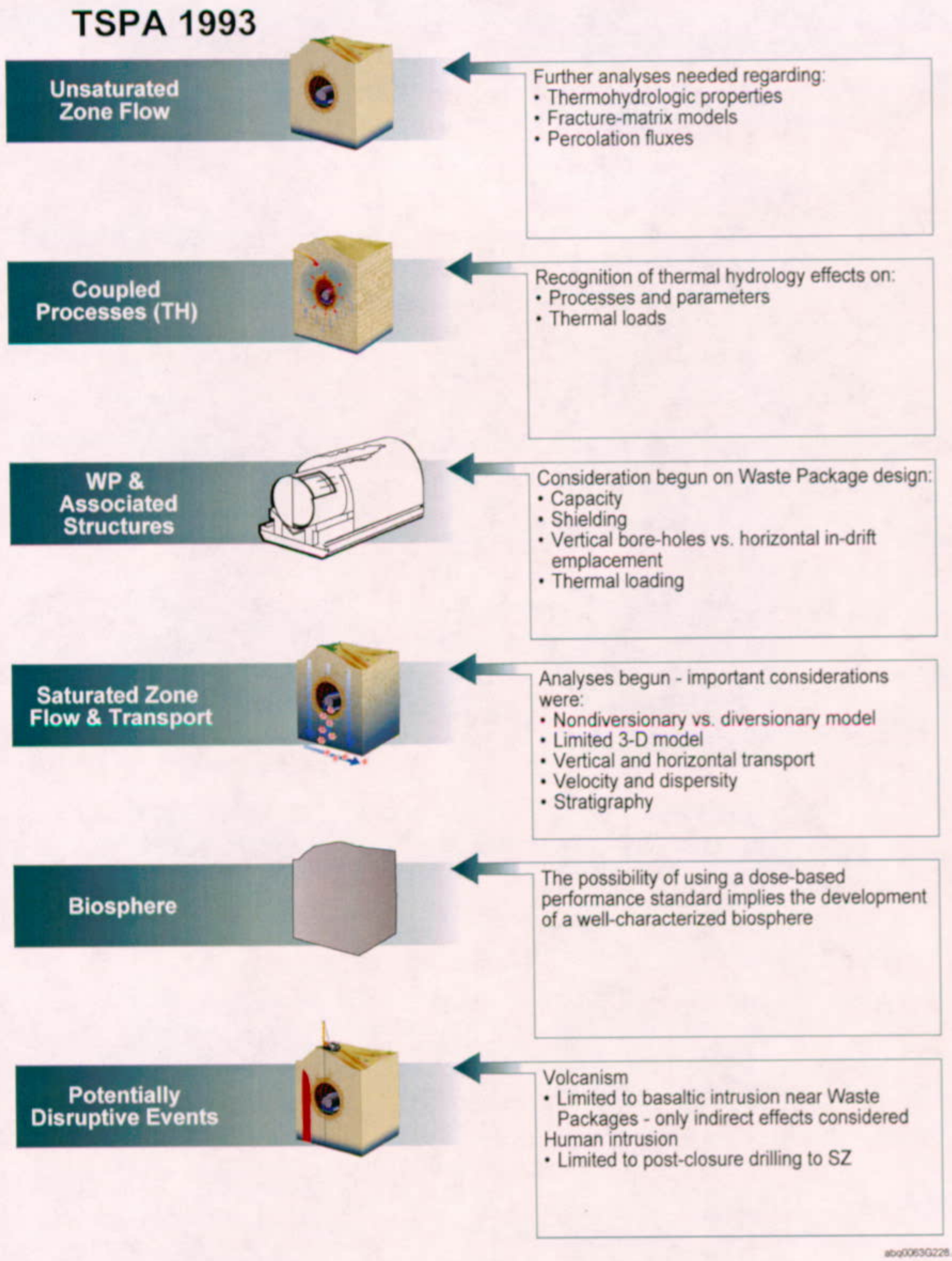
NOTE: TH – thermal hydrology; WP – waste package

Figure 1.5-2. Subsystem Model Abstractions Available for the 1991 Total System Performance Assessment



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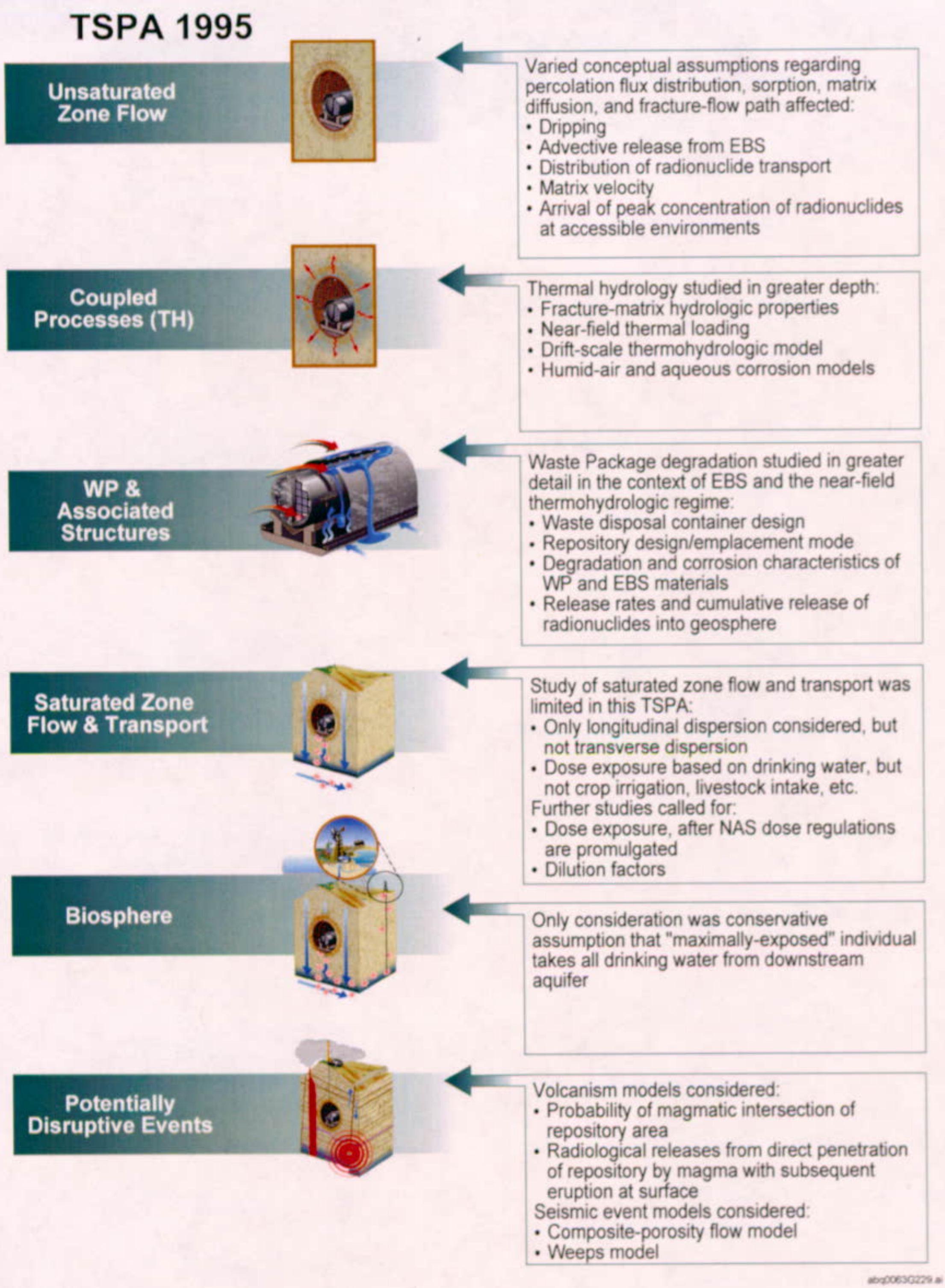
NOTE: TH – thermal hydrology; WP – waste package

Figure 1.5-3. Subsystem Model Abstractions Available for the 1993 Total System Performance Assessment



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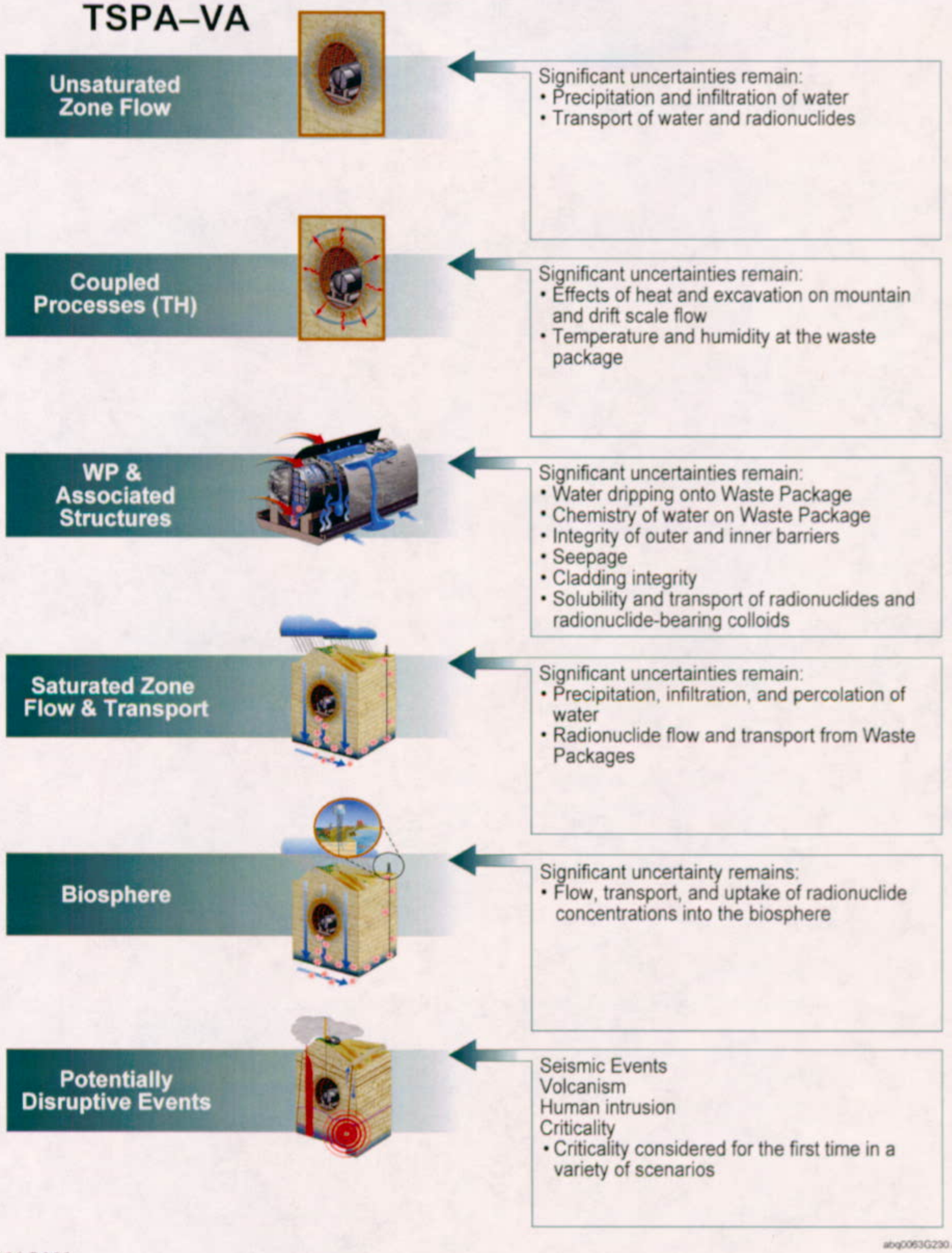
NOTE: TH – thermal hydrology; WP – waste package

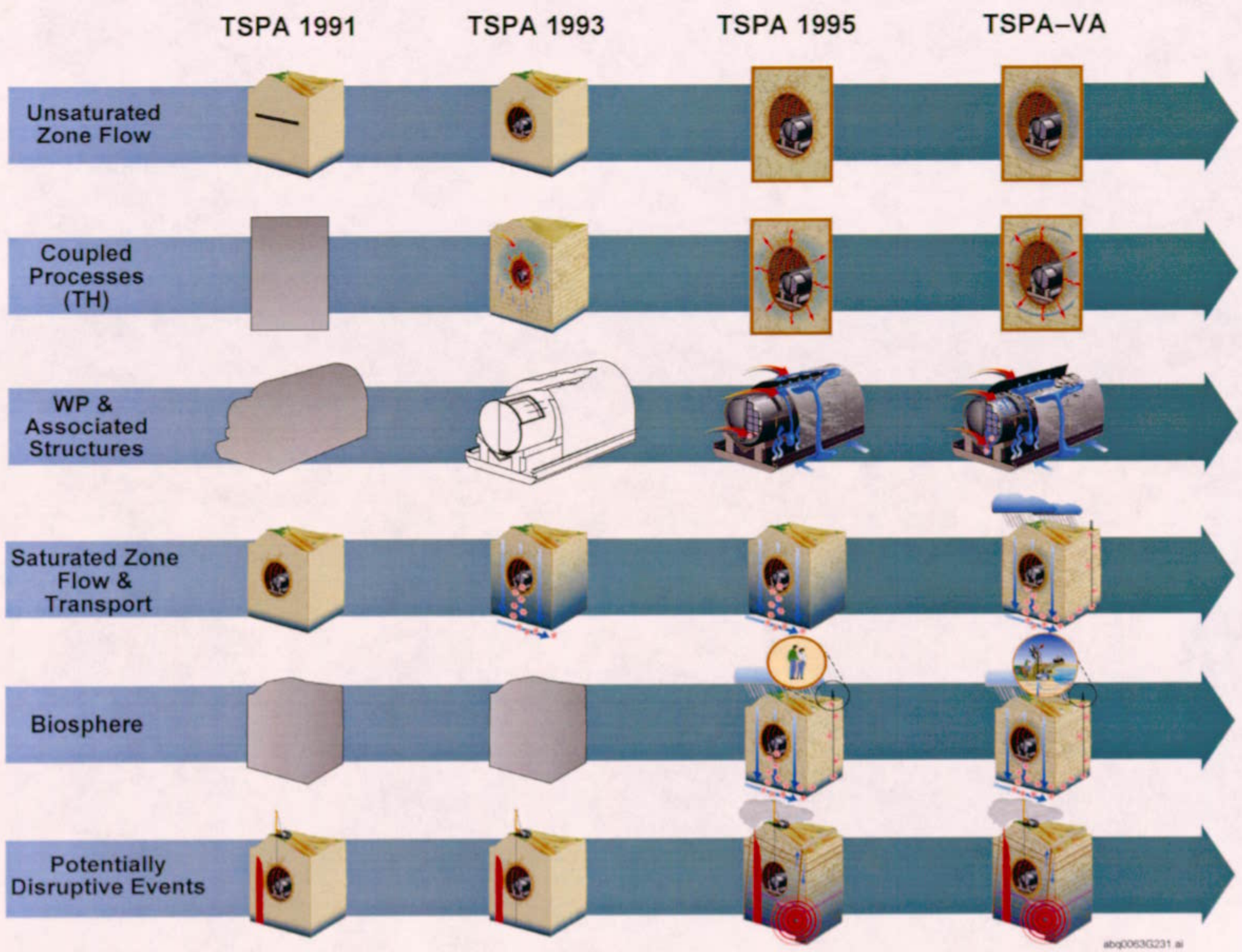
Figure 1.5-4. Subsystem Model Abstractions Available for the 1995 Total System Performance Assessment



abq0063G230

NOTE: TH – thermal hydrology; WP – waste package

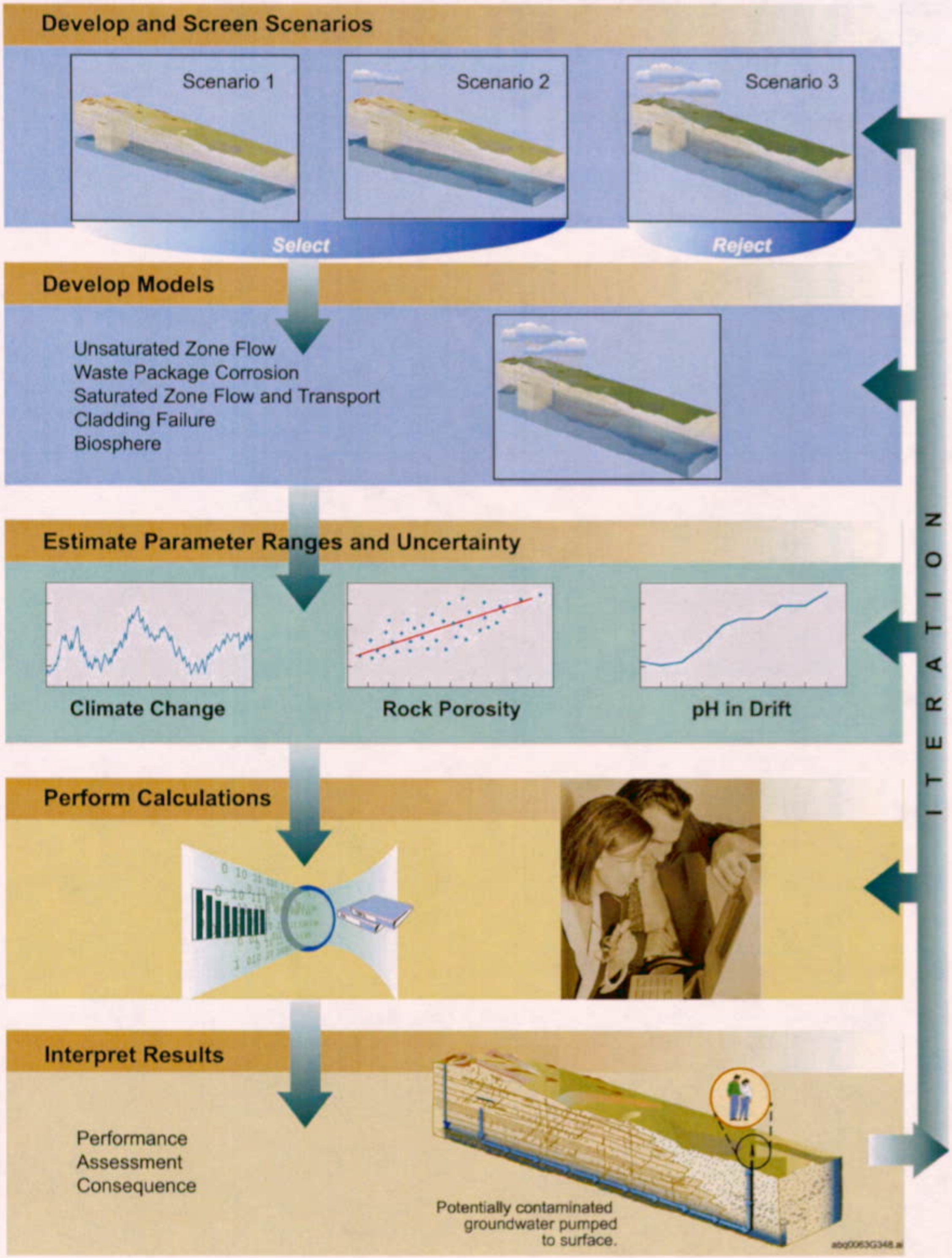
Figure 1.5-5. Subsystem Model Abstractions Available for the Viability Assessment Total System Performance Assessment



abq0063G231

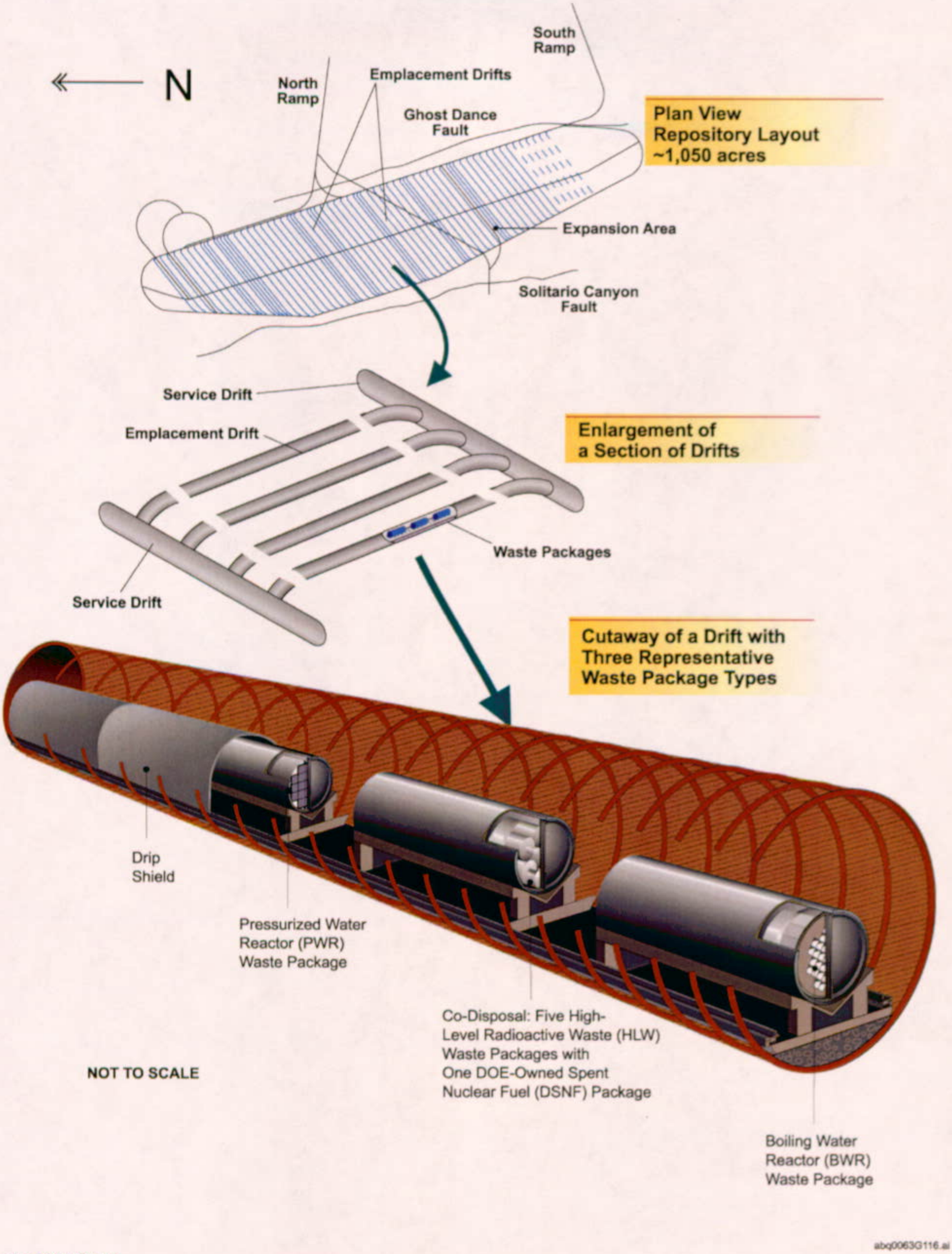
NOTE: TSPA-VA – Total System Performance Assessment – Viability Assessment; TH – thermal hydrology; WP – waste package.

Figure 1.5-6. Increase in Sophistication of Subsystem Component Models with Successive Iterations of Total System Performance Assessment



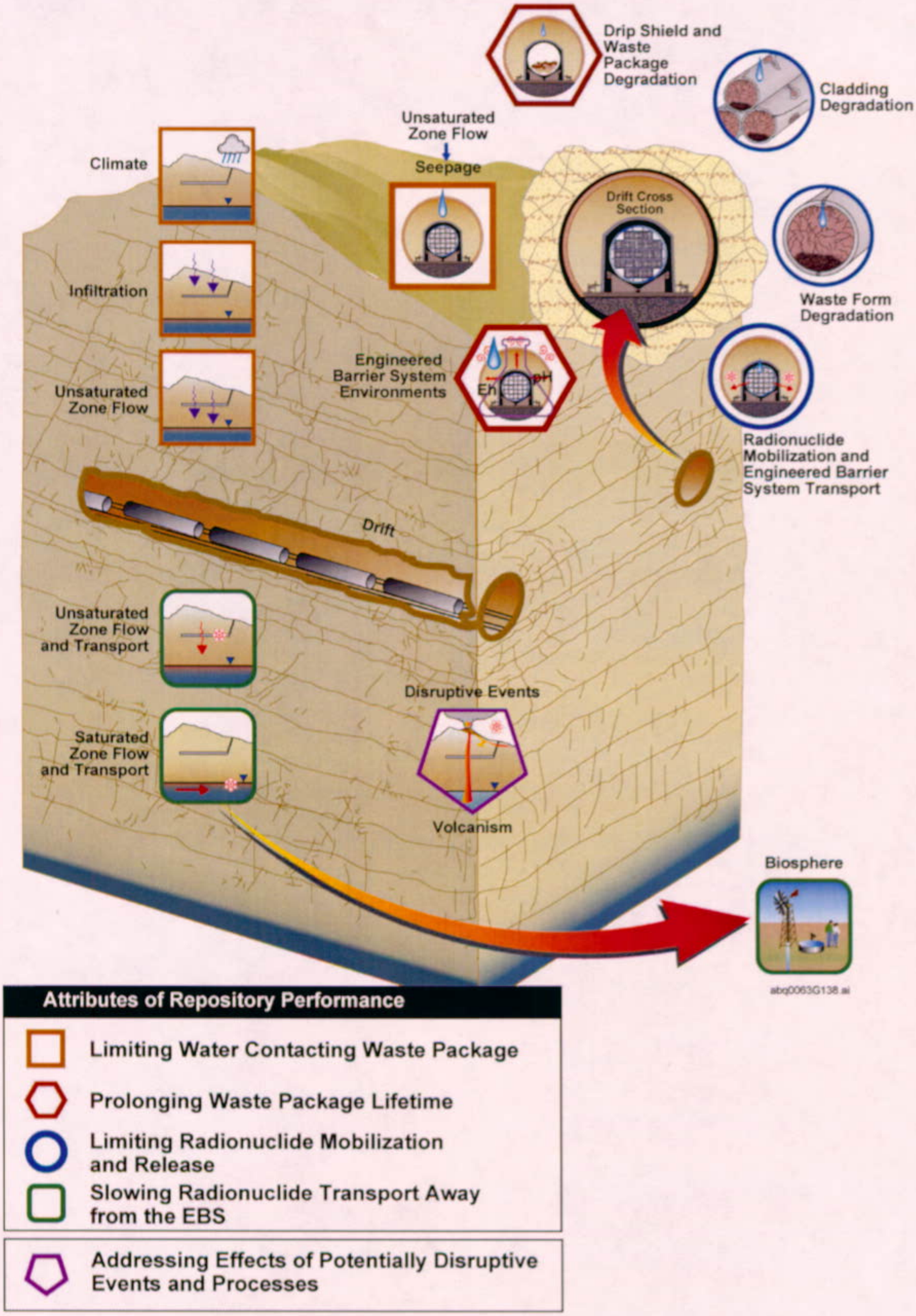
abq0063G348

Figure 1.6-1. Major Steps in a Generic Performance Assessment



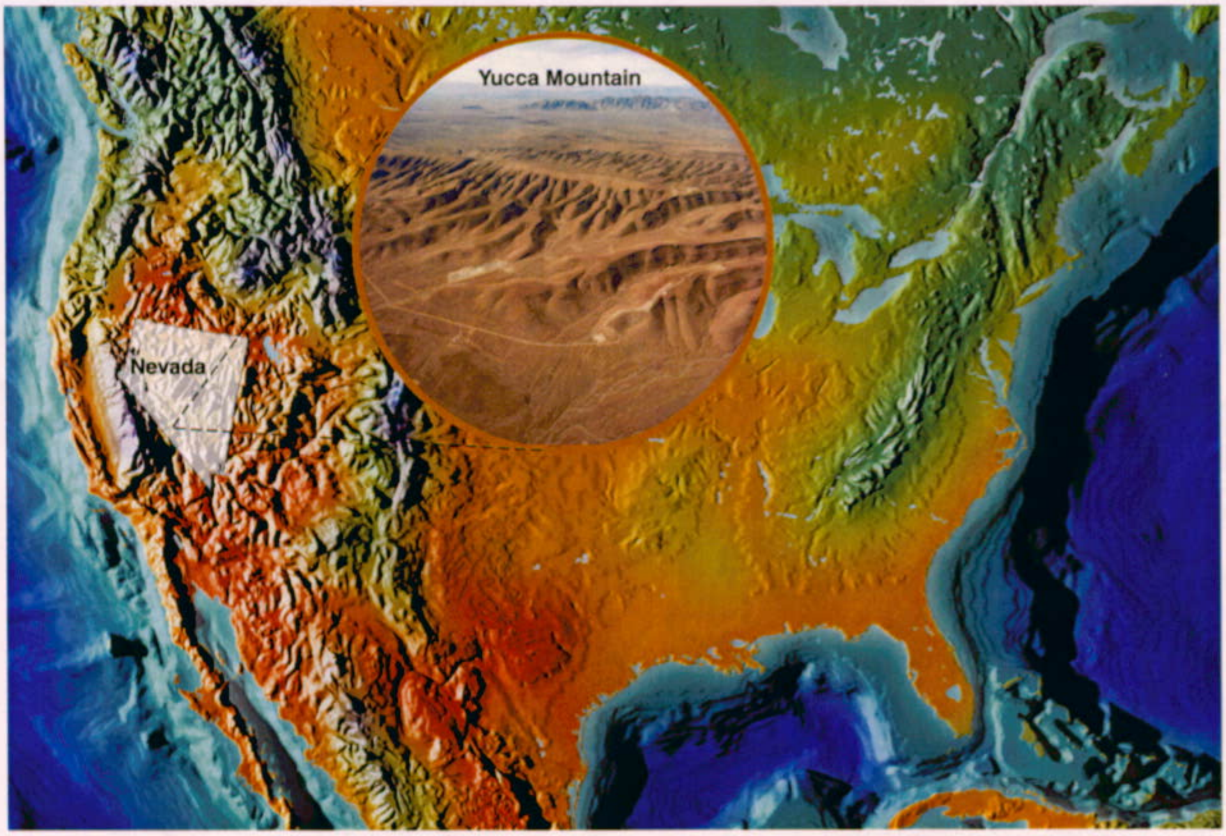
abq0063G116

Figure 1.7-1. Base Case Repository Design at Time of Closure



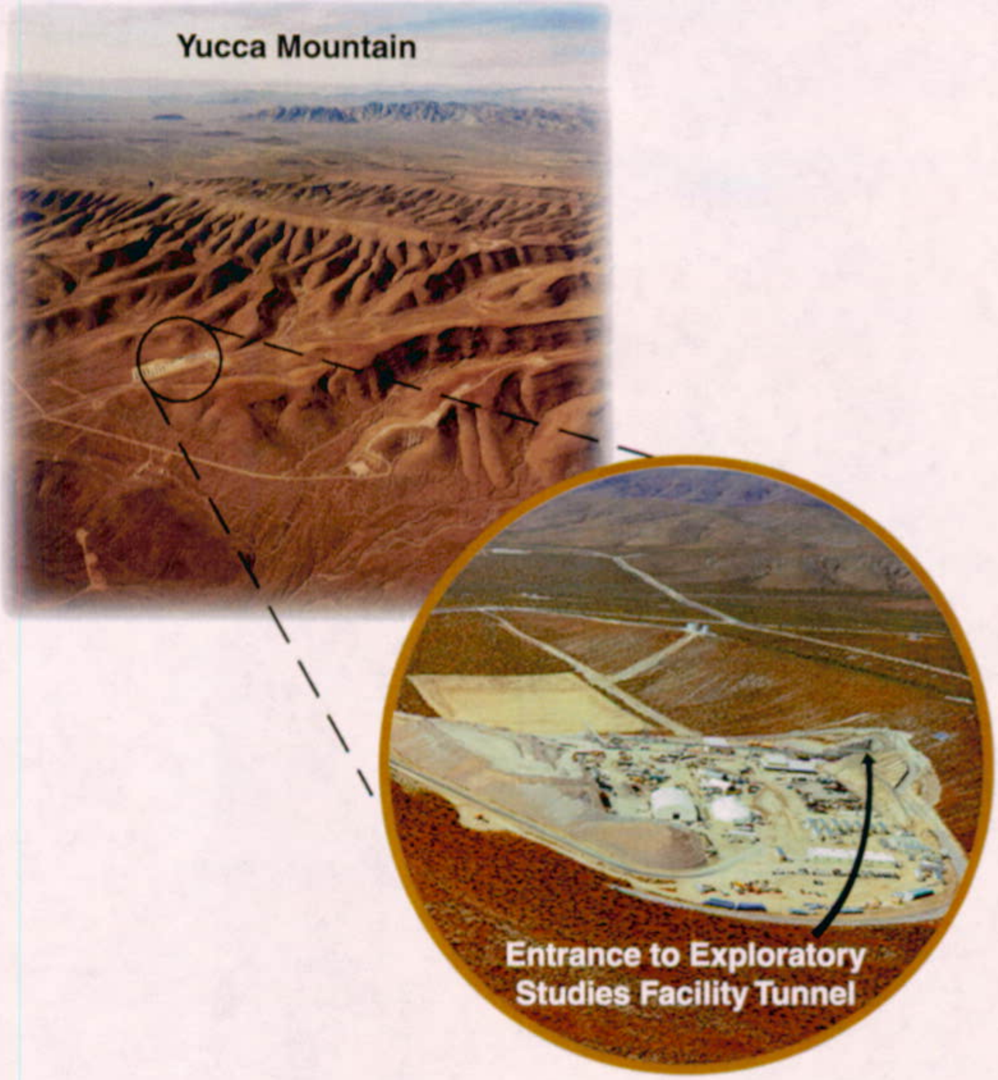
abq0063G138

Figure 1.8-1. Attributes of Repository Performance



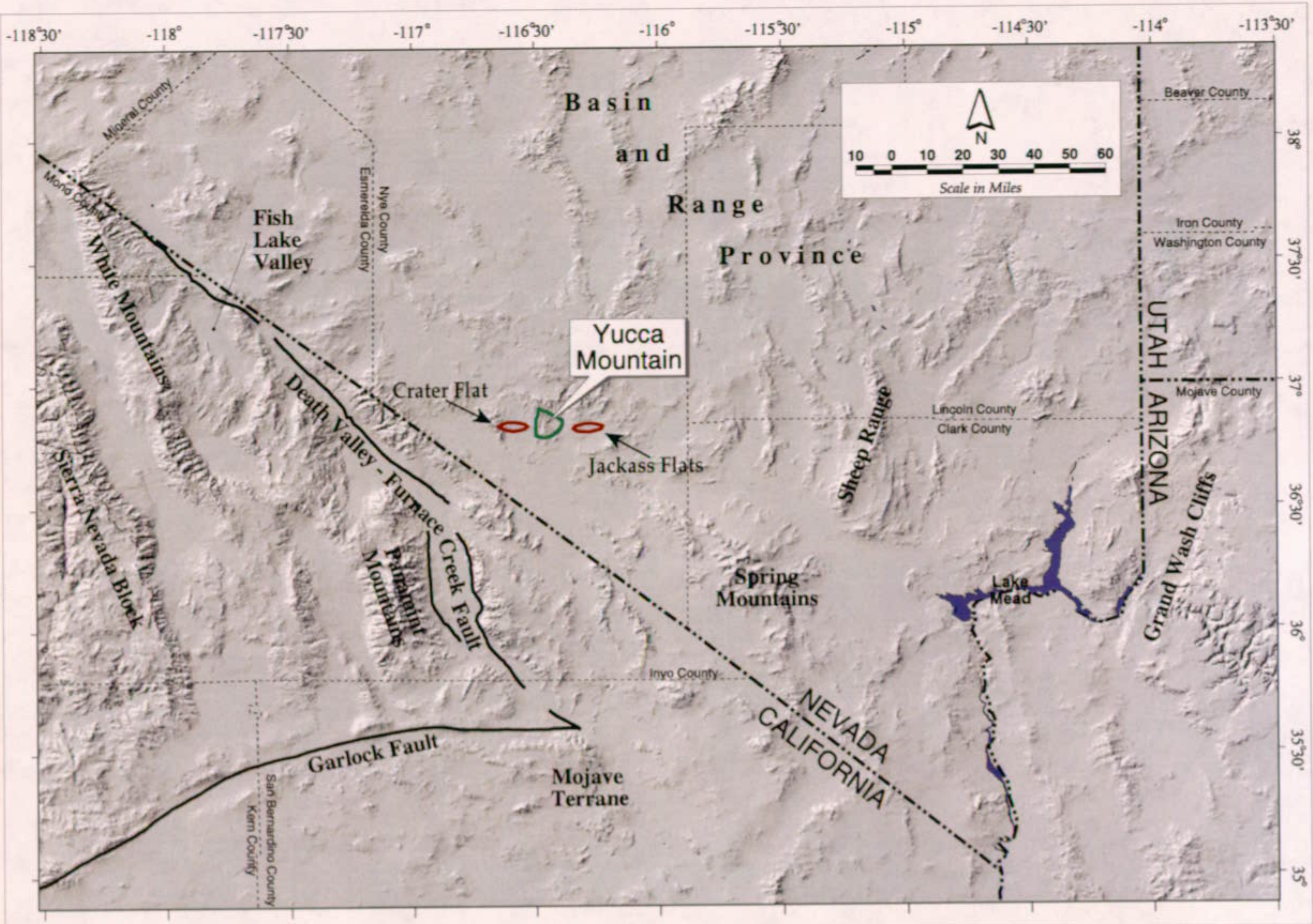
abq0063G416

Figure 1.8-2. Location of Yucca Mountain



abq0063G072

Figure 1.8-3. Yucca Mountain Area

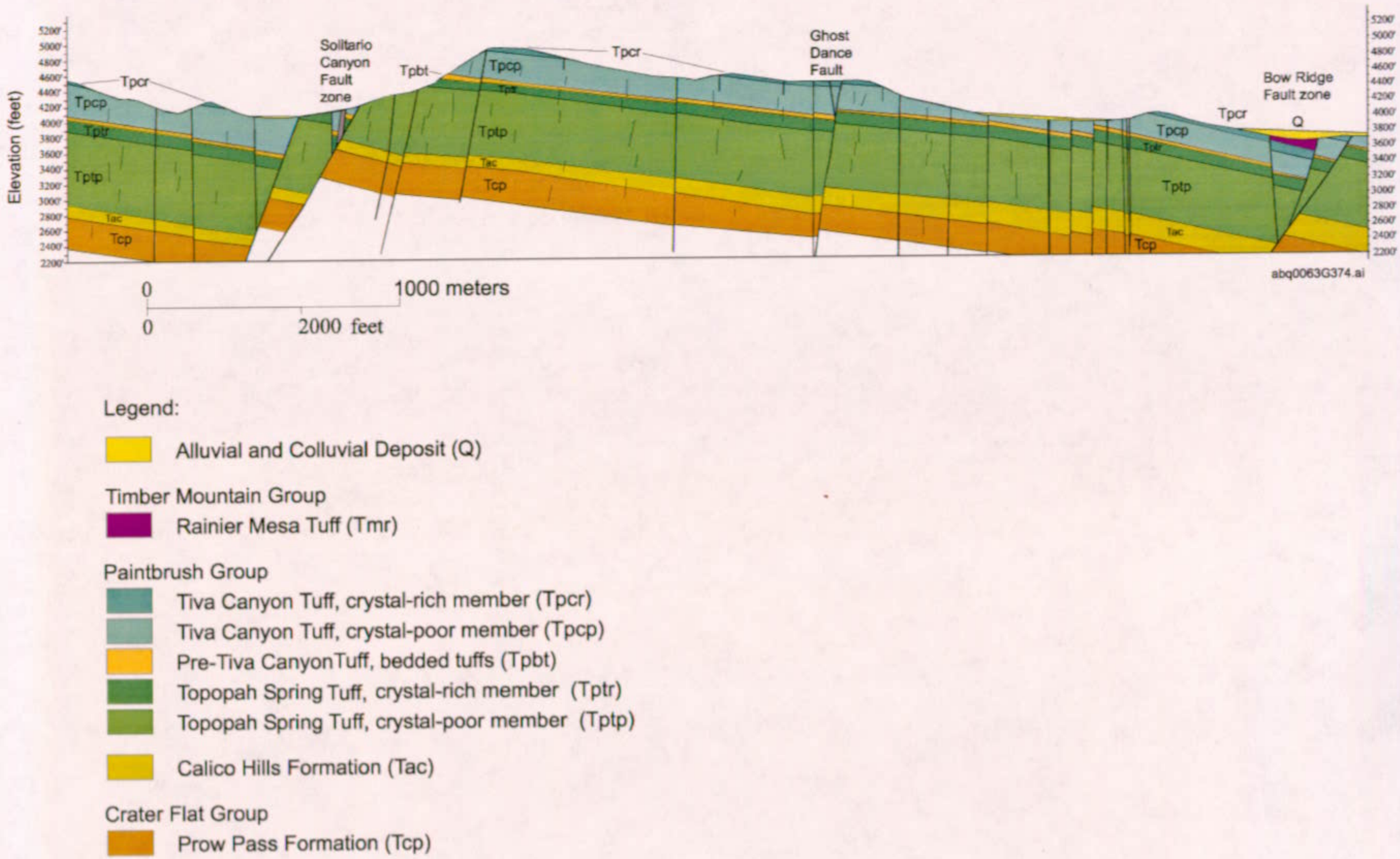


Note: Only faults with the highest rates of activity are shown

abq0063G372

Source: DOE 1998 [100548], Figure 2-11

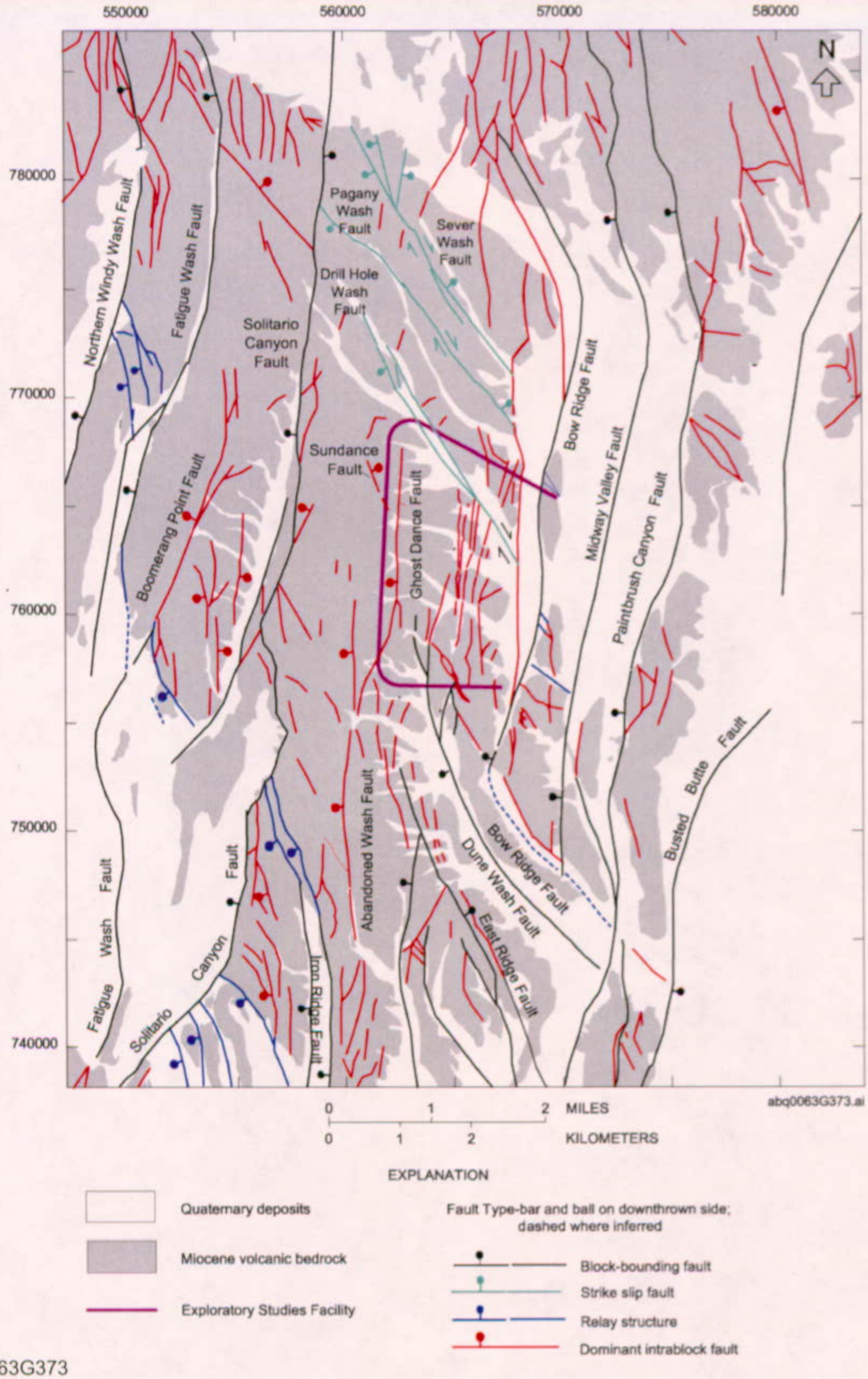
Figure 1.8-4. Physiographic Map of the Southern Basin and Range Province



abq0063G374

Source: Day et al. 1997 [100133], Plate 1

Figure 1.8-5. Generalized Cross Section through Yucca Mountain



abq0063G373

Source: Day et al. 1997 [100133], Figure 3

Figure 1.8-6. Location of Faults at Yucca Mountain

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2. YUCCA MOUNTAIN SITE CHARACTERIZATION PROJECT TOTAL SYSTEM PERFORMANCE ASSESSMENT FOR THE SITE RECOMMENDATION

The concept of TSPA and the generic TSPA process are described in Section 1 of this report. The acceptability of TSPA as a tool for analyzing a nuclear waste repository system is also described in that section. Section 2 of this report discusses the more specific use of the TSPA process for Yucca Mountain, including the approach and specific method.

Section 1.2 discusses the objectives of the TSPA-SR. The primary objective was to evaluate the performance of the system in support of the SR. To accomplish this objective, available scientific information about the natural geologic setting was used in the TSPA, along with data about the engineered components and their interactions with the geologic setting.

The primary performance measure used for evaluation of system performance is dose to the average member of the critical group at 20 km from the potential repository boundary. The time scale of regulatory concern for the analysis was primarily 10,000 years, although the evaluation was extended to fully consider some of the processes affecting release of radionuclides from the system.

Section 2.1 contains a general discussion of the approach used to analyze the potential Yucca Mountain repository system in the TSPA-SR. Building the capability for an integrated system analysis requires input from many individual disciplines. In addition, the analyses benefited from comments received from both internal and external reviewers of previous system analyses conducted for the potential repository. Sources of data and information for constructing the TSPA-SR include previous TSPAs; documented models of the principal components; workshops to abstract, or simplify, the process model components; expert elicitations of some model components; and external reviews. Nine principal components in the TSPA models are combined to evaluate the overall potential repository system. The presence of water is a key factor in initiating release of radionuclides from waste packages and moving them through the environment to a contact point with humans. For this reason, evaluation of the model components focused on the ability either to keep water from contacting the waste or to minimize releases of radioactivity from the potential repository if waste packages are breached.

Section 2.2 provides a detailed description of the method used to analyze the potential repository system in the TSPA. While Section 2.1 discusses the process model components as individual models, Section 2.2 provides a road map for reassembling, or coupling, the component models into one integral whole. Section 2.2 presents an overview of the TSPA-SR method for mathematical and numerical modeling of the individual processes, including their uncertainty, and the approach for combining them into an overall model and computer code. It includes discussions about information flow between the models and the architecture of the computer program for the TSPA-SR that facilitates the information flow. It also discusses the method for presenting the key results; in particular, it discusses how to show the influence of uncertainty of inputs on performance estimations and the effect of uncertainty on the base case.

2.1 TOTAL SYSTEM PERFORMANCE ASSESSMENT APPROACH

2.1.1 Development of an Integrated Total System Performance Assessment Approach

The potential Yucca Mountain repository system is a combination of integrated processes that can be summarized in 1 basic objective, 4 attributes, and 25 factors for the nominal case. An additional attribute and 8 factors are included as disruptive events and processes. The basic objective of the waste disposal system is to contain and isolate radioactive waste so that the dose impact to humans is attenuated to a relatively benign level. This objective manifests itself in the following attributes of nominal repository performance:

- Limiting water contacting the waste packages
- Prolonging waste package lifetime
- Limiting radionuclide mobilization and release from the EBS
- Slow transport away from the EBS.

The attribute of disruptive repository performance is:

- Low mean annual dose even considering effects of potentially disruptive processes and events.

Table 2.1-1 shows these attributes and the factors associated with them, as well as the TSPA-SR model components that are pertinent to the nominal and disruptive factors.

Building an integrated system analysis capability requires input from the many disciplines that compose the system. The construction of the model also benefits from comments from internal and external reviewers of previous system analyses conducted for the potential repository. The analyses documented in this volume have benefited from such interactions. The final approach and methods used in the analyses have evolved following completion of the *Total System Performance Assessment-Site Recommendation Methods and Assumptions* (CRWMS M&O 2000 [147323]). The major sources of information that form the bases for the development of the FEPs; subsequent methods; assumptions; and component models used in the TSPA-SR documented here are illustrated in Figure 2.1-1. The major sources for development of FEPs and other information required in the TSPA-SR include the following:

- DOE and non-DOE TSPAs of Yucca Mountain

- Documented models and analyses describing each of the principal components of the TSPA

- Workshops on abstractions of individual process model components used in the TSPA

- Reviews of the TSPA plans, methods, and assumptions

- NRC IRSRs which address the status of key technical issues assessed in the TSPA

- Expert elicitations of key process model components used in the TSPA.

Table 2.1-1. Factors Affecting Expected Postclosure Performance for the Site Recommendation and Their Corresponding Total System Performance Assessment-Site Recommendation Model Components

Attributes of Repository Performance	Factors[a]	TSPA Model Components
Limiting water contacting waste packages	Climate	UZ Flow
	Net Infiltration	
	UZ Flow	
	Coupled Effects on UZ Flow	
	Seepage into Emplacement Drifts	Seepage
	Coupled Effects on Seepage	
	In-Drift Physical and Chemical Environments (Environments on Drip Shield; Environments on Waste Package)	EBS Environments
	In-Drift Moisture Distribution (Moisture on Drip Shield; Moisture on Waste Package)	
Prolonging waste package lifetime	Drip Shield Degradation and Performance	Waste Package Degradation
	Waste Package Degradation and Performance	
Limiting radionuclide mobilization and release from the EBS	Radionuclide Inventory and Distribution in Repository	Waste Form Degradation Radionuclide Mobilization and EBS Transport
	In-Package Environments	
	Cladding Degradation and Performance	
	CSNF Degradation and Performance	
	DSNF Degradation and Performance	
	DHLW Degradation and Performance	
	Dissolved Radionuclide Concentrations	
	Colloid-Associated Radionuclide Concentrations	
	In-Package Radionuclide Transport	
	EBS (Invert) Degradation and Performance	
Slow radionuclide transport away from the EBS packages	UZ Radionuclide Transport (Advective Pathways, Retardation, Dispersion)	UZ Transport
	Coupled Effects on UZ Radionuclide Transport	
	SZ Radionuclide Transport (Advective Pathways, Retardation, Dispersion)	SZ Flow and Transport
	Wellhead Dilution	
	Biosphere Dose Conversion Factors	Biosphere Transport and Uptake
Low mean annual dose even considering effects of potentially disruptive processes and events	Probability of Volcanic Eruption	Volcanic Eruption
	Characteristics of Volcanic Eruption	
	Effects of Volcanic Eruption	
	Atmospheric Transport of Volcanic Eruption	
	Biosphere Dose Conversion for Volcanic Eruption	
	Probability of Igneous Intrusion	Igneous Intrusion
	Characteristics of Igneous Intrusion	
	Effects of Igneous Intrusion	

Source: [a] Modified after *Repository Safety Strategy Revision 4* (CRWMS M&O 2000 [148713])

Each of these sources is described in the following paragraphs. In addition, indirect information derived from other radioactive and nonradioactive waste programs has been used in the development of the approach and methodology used in the TSPA-SR. The detailed technical basis for the TSPA is documented in the *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]).

DOE contractors completed previous TSPA analyses in 1991, 1993, 1995, and 1998. These analyses are documented in *TSPA 1991: An Initial Total-System Performance Assessment for Yucca Mountain* (Barnard et al. 1992 [100309]), *Total-System Performance Assessment for Yucca Mountain - SNL Second Iteration (TSPA-1993)* (Wilson et al. 1994 [100191]), *Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository* (CRWMS M&O 1995 [100198]), and *Total System Performance Assessment*, Volume 3, of *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998 [100550]). The general objective of these scoping analyses was to refine the methods and approach that would be applied in the development of the Site Recommendation and, ultimately, in the License Application. An additional objective was to assist DOE in prioritizing design and scientific investigations on the key components that most significantly impact performance. The knowledge gained in these analyses has assisted DOE in identifying which components significantly influence the performance of the potential repository system and has aided in prioritizing the data gathering activities needed to support the development of defensible models of these components. These TSPA analyses are described in more detail in Section 1.5.

Other TSPA analyses not sponsored by DOE have been conducted by the Electric Power Research Institute and the NRC. The Electric Power Research Institute's most recent iteration of a TSPA is documented in *Alternative Approaches for Assessing the Performance and Suitability of Yucca Mountain for Spent Fuel Disposal* (McGuire et al. 1998 [152193]). The NRC conducted an iterative performance assessment (NRC 1999 [152183]) in parallel with the DOE TSPA conducted for the VA. Each iterative analysis of total system performance, whether performed by DOE and its contractors, NRC and its contractors, or the Electric Power Research Institute and its contractors, leads to incorporation of current lab and field information and improved insights about the performance of the potential repository system. In addition, a review of the conceptual models and parameters used in the analyses provides a basis for defining the significance of the range of uncertainties examined. In general, all of these analyses converge on a few key components analogous to the factors identified in Table 2.1-1. These factors are reflected in the NRC key technical issues and the Issue Resolution Status Reports, which address portions of the key technical issues.

TSPAs are based on a number of building blocks. Principal among these are models that describe how Yucca Mountain behaves in the presence of a repository and how the engineered system behaves within the environmental setting. These process models are designed to describe the behavior of individual and coupled physical and chemical processes. A significant portion of the DOE site characterization program has been aimed at developing the scientific bases for the most reasonable representation of the Yucca Mountain site and its associated engineered barriers. Those scientific bases serve as the foundation for the process models used in the TSPA. The basis for these models is described in more detail in the process model reports, and their use in the TSPA is discussed in Section 3.

To ensure that this TSPA would be based on the most current scientific knowledge, a series of workshops were held in 1998 and 1999 to bring together YMP scientists, engineers, and performance assessment analysts. These individuals consisted of scientists and engineers from the DOE, national laboratories, the U.S. Geological Survey, and the Civilian Radioactive Waste Management System Managing and Operating Contractor. Observers at these workshops included technical staff from such regulatory agencies as the NRC and the EPA and their contractors, along with external oversight groups, such as the Nuclear Waste Technical Review Board. The aim of these workshops was to develop a strategy for identifying and incorporating the relevant aspects of the individual process models into the TSPA analyses. In addition to defining the appropriate approach for abstracting the important elements of each process model, these workshops assisted the DOE in defining and prioritizing the major technical issues that had to be addressed in the TSPA. Many of these issues correspond directly with the key technical issues raised by the NRC. Each workshop culminated in a plan for incorporating that component in the TSPA and linking it to other portions of the TSPA. These plans were summarized in *Total System Performance Assessment-Site Recommendation Methods and Assumptions* (CRWMS M&O 2000 [147323]).

Acknowledging diverse technical and scientific opinions is an important part of any engineering and scientific endeavor that is as complex as analyzing the design and performance of a potential disposal facility for high-level radioactive waste. For the VA, the DOE sponsored 5 expert elicitations on key process models for the TSPA. The goal of these elicitations was to solicit the judgment of nationally and internationally recognized scientists in quantifying the uncertainty associated with each of these process models and the uncertainty in the parameter values used in those models. The process followed the procedures and approaches for eliciting expert judgments that have been formalized in documents like the DOE guidance for the formal use of expert judgment (YMP 1995 [100381]) and the NRC Branch Technical Position on the use of expert elicitation in the high-level radioactive waste program (Kotra et al. 1996 [100909]). The assessments and probability distributions from these elicitations provide a reasonable aggregate representation of the knowledge and uncertainties about issues in evaluating the various processes important to potential repository performance. The five areas evaluated in the elicitations were:

- UZ flow (CRWMS M&O 1997 [100335])
- Waste package degradation (CRWMS M&O 1998 [100349])
- SZ flow and radionuclide transport (CRWMS M&O 1998 [100353])
- Near-field environment (CRWMS M&O 1998 [100351])
- Waste form degradation (CRWMS M&O 1998 [100374]).

In addition to these elicitations, the DOE conducted external elicitations of the potential hazards associated with either volcanically or seismically induced events. The use of these results in the analysis of the potential effects of disruptive scenarios is described in Section 3.10. Following the VA, additional work was conducted to eliminate the need for utilization of the results of these expert elicitations in TSPA-SR, with the exception of the SZ expert elicitation results, potential volcanic hazard analysis and potential seismic hazard analysis.

In addition to DOE-sponsored development of the TSPA models, several external oversight groups provided input throughout the process of defining and implementing the approach and methods. These groups include the NRC and its contractor, the Center for Nuclear Waste Regulatory Analysis; the NRC Advisory Committee for Nuclear Waste; and the congressionally chartered Nuclear Waste Technical Review Board. These organizations have published a range of technical comments on the TSPA and conducted numerous briefings over the last two years on the progress of the TSPA. Their comments have aided in defining the most appropriate means of describing and analyzing the performance of the Yucca Mountain site and the engineered barriers associated with the reference design and design options.

The approach to developing the TSPA-SR models, as noted above, included the development of FEPs, the screening of these FEPs, and construction of scenario classes, which are groupings of closely related FEPs and scenarios that have been combined for the purpose of assigning probabilities and screening, consistent with guidance provided by the NRC (NRC 2000 [149372], Section 4.2.4, 4.2.5). This FEPs approach is discussed in Section 2.1.1.1.

2.1.1.1 Implementation of the Features, Events, and Processes Approach

A series of DOE-sponsored TSPA scoping analyses were performed between 1991 and 1998 (see Section 1.5). During the iterative process of performing these analyses, knowledge was gained regarding (1) the TSPA methods and approach, (2) key features of the potential Yucca Mountain repository system, and (3) events and processes that most significantly impact postclosure performance. Additional information about the important FEPs relevant to Yucca Mountain was collected from non-DOE-sponsored TSPAs, process model workshops and analyses, NRC IRSRs, expert elicitations, and external oversight groups (see Section 2.1.1). The collective results of these activities, which represent an informal approach to identifying the FEPs and scenarios important to the postclosure performance of the potential Yucca Mountain repository system, will provide input to successive iterations of the Repository Safety Strategy (CRWMS M&O 2000 [148713]). This informal FEPs approach identified 5 attributes and 33 factors important to postclosure performance (Table 2.1-1). The approach also identified a nominal scenario class (4 attributes and 25 factors), a disruptive event scenario class (1 attribute and 8 factors), and associated model components.

Under the provisions of the DOE's Interim Guidance (Dyer 1999 [105655], Section 102[j]), a performance assessment is defined as a systematic analysis that (1) identifies the FEPs that might affect the performance of the potential geologic repository, (2) examines the effects of such FEPs on the performance of the geologic potential repository, and (3) estimates the expected annual dose to a specified receptor group. The performance assessment must also provide the technical basis for inclusion or exclusion of specific FEPs in the performance assessment (Dyer 1999 [105655], Section 114). To address these requirements, a formal approach to selecting scenario classes for analysis in the TSPA-SR was implemented, based on the identification and screening of FEPs potentially relevant to the postclosure performance of the potential repository. The formal FEPs approach builds from the attributes, factors, scenarios, and model components identified in the previous informal approach.

The formal FEPs and scenario development process adopted for the TSPA-SR is based on the methodology developed by the NRC (Cranwell et al. 1990 [101234], Section 2.0). The approach is fundamentally the same as that used in many performance assessments, including a recent analysis of the potential Yucca Mountain repository by the NRC (Wescott et al. 1995 [100476]);

it has also been used by the DOE for examining the Waste Isolation Pilot Plant (DOE 1996 [100975], Section 6.2), by the Nuclear Energy Agency (1992 [100479]), and by other international radioactive waste programs (e.g., Skagius and Wingefors 1992 [101018], Section 2).

The five principal steps in the scenario development process are illustrated in Figure 2.1-2, discussed in detail in subsequent subsections, and outlined below:

1. Identify FEPs potentially relevant to the long-term performance of the disposal system.

2. Classify the FEPs to support evaluation of completeness and to facilitate screening.

3. Screen the FEPs using defined criteria to identify those that should be included in the TSPA analysis and those that can be excluded from the analysis.

4. Construct scenario classes (sets of related scenarios) from the retained (included) FEPs, as appropriate.

5. Screen the scenarios classes using the same criteria applied to the FEPs to identify any scenario classes or scenarios that can be excluded from the TSPA analysis.

These 5 steps differ slightly from those identified in the *Total System Performance Assessment-Site Recommendation Methods and Assumptions* (CRWMS M&O 2000 [147323], Section 2.2). The changes were made so the 5 steps of the TSPA-SR approach would correspond directly to the 5 elements of scenario analysis acceptance criteria outlined in the *TSPA&I IRSR* (NRC 2000 [149372], Section 4.2).

A YMP FEP database was developed to catalog the following information: a comprehensive list of FEPs that have the potential to influence repository performance, a systematic classification structure for FEPs that helps to evaluate completeness and facilitate screening, and screening information that summarizes the technical basis for inclusion or exclusion of each FEP in the TSPA-SR analyses. The YMP FEP Database serves as a communication tool for FEPs information. The information contained in the database was generated in separate Analysis Model Reports. The YMP FEP database REV00 (the current version) and documentation of its development are contained in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806]). The database documentation is also summarized in Appendix B of this report.

Step 1: Identification of FEPs–The development of a comprehensive list of FEP potentially relevant to the postclosure performance of the potential Yucca Mountain repository is an ongoing, iterative process based on site-specific information, design, and regulations. The YMP FEP list was initially developed from a comprehensive list of FEPs from other international radioactive waste disposal programs (see Appendix B of this report and CRWMS M&O 2000 [150806], Section 2.1) and was supplemented with additional YMP-specific FEPs from project literature, technical workshops, and reviews (see Appendix B of this report and CRWMS M&O 2000 [150806], Sections 2.2 through 2.4). The YMP FEP list is open and may continue to expand if additional FEPs are identified during the Site Recommendation and License Application processes.

The sources identified above produced 1,646 specific FEPs. These FEPs, when combined with the 151 general FEP classifications (described in Step 2 below), result in a YMP FEP list that contains 1,797 entries.

The NRC acceptance criteria for the identification of FEPs (NRC 2000 [149372], Section 4.2.1) address the comprehensiveness of the FEP list. Absolute proof of comprehensiveness is not possible. However, the comprehensiveness of the YMP FEP list derives from (a) the diverse backgrounds of the international waste disposal programs contributing to the list, (b) the variety of methods used to identify FEPs including expert judgment, informal elicitation, event tree analysis, stakeholder review, and regulatory stipulation, (c) the iterative discussions and reviews (i.e., at technical workshops and in analysis model reports) of important YMP attributes, factors, and model components, (d) the systematic and comprehensive classification structure (as described in Step 2 below) that ensures that no relevant subject area is overlooked, and (e) the fact that FEPs cannot be removed from the list, they can only be screened out (excluded) from the analysis.

Step 2: Classification of FEPs–The all-inclusive approach used to identify FEPs (described in Step 1 above) produced a large number (1,646) of specific FEPs, and resulted in considerable redundancy in the FEP list, because the same FEPs were frequently identified by multiple sources. To better organize these FEPs and to help evaluate the completeness of the FEP list, a hierarchical classification structure was adopted within the YMP FEP database (see Appendix B of this report and CRWMS M&O 2000 [150806], Section 3.1). The FEP classification structure was defined by 4 high-level layers, 12 associated categories, and 135 underlying headings. Each of the 1,646 specific FEPs identified in Step 1 was assigned (mapped) to a single heading in the YMP FEP database. This mapping resulted in a classification where all related FEPs were grouped together under the same classification heading (with additional relationships to the overarching categories and levels). The implementation of the classification structure in the database produced a YMP FEP list containing 1,797 entries, composed of the 1,646 specific FEPs and the 151 classification (layer, category, and heading) entries.

To eliminate the redundancy and to create a more efficient aggregation of FEPs to carry forward into the screening process (described in Step 3 below), each of the 1,797 entries in the YMP FEP database was further identified as either a primary, secondary, or classification (layer, category, or heading) entry. The process and criteria for assigning FEPs to one of these categories is described in Appendix B of this report and in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806], Section 3.2).

Primary FEPs encompass a single process or event, or a few closely related or coupled processes or events that can be addressed by a specific screening discussion. A primary FEP may also include one or more related secondary FEPs that are covered by the same screening discussion. Secondary FEPs are (1) redundant to another FEP (e.g., several international programs identified the same FEP), (2) specific to another program (and captured more generally in a different YMP-specific FEP), or (3) better captured or subsumed in another similar but more broadly defined YMP-specific FEP. Each secondary FEP was mapped to a primary FEP and was completely addressed by the screening discussion of that primary FEP.

Classification entries represent the hierarchical levels of classification within the database. They are defined too broadly to be addressed by a single screening discussion (as with a primary FEP) and cannot be encompassed by an overlying FEP (as with a secondary FEP). Rather, they classify one or more underlying related primary FEPs and do not require screening discussions. If the FEPs grouped under a specific heading entry were closely enough related that they could all be addressed by a screening discussion at the overlying heading level, the heading entry (which would otherwise be defined as a classification entry) was designated as a primary FEP. The underlying FEPs were designated as secondary FEPs to the heading-level primary FEP.

The classification approach described in this Step resulted in 111 classification entries (151 less 40 heading entries that were reclassified as primary FEPs), 323 primary FEP entries (including the 40 headings) and 1,363 secondary FEP entries. The classification approach was designed to produce a set of primary FEP entries that capture all of the issues relevant to the postclosure performance of the potential Yucca Mountain repository. Therefore, it was only necessary to carry forward the 323 primary FEPs for screening (as described in Step 3 below). Screening of the secondary (and classification) FEPs was not required because the aspects of the secondary FEPs were encompassed by the primary FEPs.

The NRC acceptance criteria for the classification of FEPs (NRC 2000 [149372], Section 4.2.2) address the grouping and categorization of FEPs. NRC guidance accompanying these criteria suggests grouping potentially disruptive events into "event classes" that contain related events, to ensure that event probabilities are not underestimated by defining events too narrowly (NRC 2000 [149372], Section 4.2.2). The classification approach adopted for TSPA-SR produced an aggregated set of primary FEPs for screening that covered all identified potentially relevant Yucca Mountain FEPs. Although the DOE has not adopted the term "event class" because it is inconsistent with the proposed regulatory requirement to consider all FEPs rather than just events, the evaluation of probabilities for primary FEPs achieves the goal of assigning probabilities at an appropriately broad level of categorization. Documentation of the grouping and categorization of YMP FEPs is in Appendix B of this report and in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806], Section 3).

Step 3: Screening of FEPs–Each of the 323 primary FEPs identified in Step 2 above was screened for inclusion or exclusion in the TSPA on the basis of three criteria, developed from DOE's Interim Guidance (Dyer 1999 [105655]). The three criteria, described in Appendix B of this report and in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806], Section 4), are outlined below:

- **Regulatory**–FEPs that are inconsistent with regulatory guidance (Dyer 1999 [105655], Subpart E) may be excluded (screened out) from the TSPA analysis. The most notable examples are the regulatory specification of the human intrusion scenario and the critical group characteristics.

- **Probability**–FEPs that have less than one chance in 10,000 of occurring over 10,000 years may be excluded (screened out) from the TSPA on the basis of low probability.

- **Consequence**–FEPs whose exclusion would not significantly change the expected annual dose may be excluded (screened out) from the TSPA on the basis of low consequence.

Because DOE's Interim Guidance (Dyer 1999 [105655], Section 114) allows exclusion of FEPs on the basis of either low probability or low consequence, a FEP need not be shown to be both of low probability and low consequence to be excluded. Therefore, the order in which the criteria are applied is not essential. In practice, FEPs were screened as shown in Figure 2.1-3. Regulatory criteria were examined first, and then either probability or consequence criteria examined next at the discretion of the analyst. In some cases, one component of a FEP was included while another component of the FEP was excluded.

This application of the analyst's judgment regarding the order in which to apply the criteria does not affect the final decision. FEP that were retained on one criterion were then considered against the other. Needless work developing quantitative probability arguments for low consequence events or complex consequence models for low-probability events was prevented by allowing the analyst to choose the most appropriate criterion to apply at this step (e.g., there is no need to develop detailed models of the response of the disposal system to the impact of a large meteorite if it can be shown that this event has a probability below the regulatory cutoff).

The FEP screening was performed by subject matter experts and documented in FEP analysis model reports (listed in Appendix B). Specific guidelines for the basis of screening decisions and the content of screening documentation are outlined in Appendix B of this report and in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806], Section 4.2).

Probability screening, in general, requires some information about the magnitude of the event (e.g., the probability of meteorite impacts depends on the size of the meteorite of interest). Impacts of meteorites sufficiently large to create large craters at Yucca Mountain are much less probable than smaller impacts. Thus, meteorites large enough to affect the disposal system may be screened out on the basis of low probability, but smaller meteorite impacts that produce no significant change to the disposal system may be more appropriately screened out on the basis of low consequence. Probability screening is also sensitive to the spatial and temporal scales at which FEPs are defined (meteorite impacts are less likely in shorter time intervals and at smaller locations), and probability screening must therefore be performed at reasonably coarse scales.

Consequence screening does not necessarily require detailed calculations of the consequences (i.e., expected annual dose) (NRC 2000 [149372], Section 4.2.3). The amount of information required may vary from FEP to FEP, based on the processes and events involved. Consequence screening may rely on reasoned arguments based on literature research (e.g., consequences of many geomorphic processes, such as erosion and sedimentation, can be evaluated by considering bounding rates reported in geologic literature), TSPA sensitivity analyses, or modeling studies performed outside of the TSPA (e.g., detailed criticality analyses). Consequence screening may also be based on an intermediate performance measure (e.g., radionuclide mass release to the SZ) as long as a qualitative link to the change in expected annual dose can be demonstrated.

The NRC acceptance criteria for the screening of FEPs (NRC 2000 [149372], Section 4.2.3) address the screening criteria. Guidelines were established to ensure that the screening basis and content for each primary FEP of the screening data was sufficient to satisfy the screening criteria for low probability, low consequence, or regulatory exclusion.

Step 4: Formation of Scenario Classes–The objective of scenario development is to define a limited set of scenarios that can reasonably be analyzed quantitatively while still maintaining comprehensive coverage of the range of possible future states of the disposal system. There are an essentially infinite number of possible future states, and for scenario development to be useful, it must generate scenarios that are representative of the range of futures that are potentially relevant to the licensing of the facility.

For TSPA-SR, the term "scenario" was defined as a subset of the set of all possible futures of the disposal system that contains futures resulting from a specific combination of FEPs. At a coarser level, the term "scenario class" was adopted to refer to a set of closely related scenarios. More specifically, a scenario class is defined as a set of scenarios that share sufficient similarities that they can usefully be aggregated for the purposes of a specific analysis. Further definition of these terms is found in Appendix A. Consistent with NRC guidance (NRC 2000 [149372], Sections 4.2.4, 4.2.5), scenarios are grouped into scenario classes for TSPA-SR for the purposes of probability assignment and screening. Note, however, neither the term "scenario" nor "scenario class" is defined in the proposed regulations (64 FR 8640 [101680], Section 63.2) or in the NRC acceptance criteria (NRC 2000 [149372], Section 4.2).

The number and breadth of scenario classes depends on the resolution at which scenarios have been defined. Coarsely defined scenarios result in fewer, broad scenario classes, whereas narrowly defined scenarios result in many narrow scenario classes. In turn, the number and breadth of scenarios depends on the resolution at which FEPs have been defined. There is no uniquely correct level of detail at which to define scenario classes, scenarios, and FEPs. Decisions regarding the appropriate level of resolution for the analysis are made based on consideration of the importance of the scenarios, their effects on overall performance, and the resolution desired in the results. For efficiency, scenario classes, scenarios, and FEPs should be aggregated at the coarsest level at which a technically sound argument can be made, while still maintaining adequate detail for the purposes of the analysis.

As described at the beginning of this section, an informal approach was used to identify FEPs and scenarios based on the results of prior TSPA scoping analyses, process model workshops and analyses, NRC Issue Resolution Status Reports, expert elicitations, and external oversight groups. These prior analyses identified two scenario classes: nominal performance and disruptive performance. Formal scenario development for TSPA-SR used these scenario classes (along with the attributes, factors, and model components identified in Table 2.1-1) as a basis.

All primary FEPs not excluded (screened out) from the TSPA in Step 3 above were retained for inclusion in one or more scenario classes. Each of the retained primary FEPs was identified as either an expected FEP (EFEP) or a disruptive FEP (DFEP). EFEPs are those retained FEPs that were assumed, for the purposes of the TSPA, to have a probability of occurrence equal to 1.0 (although they may have uncertain consequences). DFEPs are those retained FEPs that have a probability of occurrence less than 1.0 (but greater than the low-probability screening threshold noted in Step 3 above). The nominal performance scenario class and the associated models were constructed to include all EFEPs. The disruptive performance scenario class and the associated models were constructed to include all DFEPs in addition to all EFEPs.

Proposed 10 CFR 963.17(b) (64 FR 67054 [124754]) identifies four disruptive events that require explicit consideration in the site suitability evaluation: volcanism, seismic events, nuclear criticality, and human intrusion. For TSPA-SR, the retained DFEPs were all associated with igneous activity (volcanism) or human intrusion. In proposed 10 CFR 963.16 (64 FR 67054 [124754]) there is a requirement for human intrusion to be evaluated in a separate performance assessment (see Section 4.4). Therefore, for TSPA-SR, the disruptive performance scenario class contained only FEPs related to igneous activity and was also referred to as the igneous disruption scenario class. Within the igneous disruption scenario class, two scenarios were identified: volcanic eruption (direct release) and igneous intrusion (indirect release via groundwater).

FEPs related to seismic damage to cladding were included in the nominal scenario class (see Section 3.5.4), rather than as a disruptive scenario. This was done primarily for pragmatic, computational reasons. The damage has no effect on performance as long as waste packages remain intact, and it was determined to be more computationally efficient to treat it as a parametric uncertainty in the nominal case as opposed to building another disruptive event scenario. Other FEPs related to seismic events (ground motion and fault displacement) and the FEPS related to nuclear criticality were screened out of TSPA-SR on the basis of low consequence (for seismicity) and low probability (for nuclear criticality). Further discussion of the treatment of these and other potentially disruptive FEPs not included in TSPA-SR is provided in Section 4.5.

The two TSPA-SR scenario classes are displayed graphically using a Latin Square scenario diagram in Figure 2.1-4. This diagram shows the probability of occurrence of each scenario class, which sum to 1.0. As discussed in Section 3.10.1, the probability of occurrence for the igneous disruption scenario class is derived from expert elicitation of the annual frequency of igneous activity in the Yucca Mountain region. The probability of the nominal scenario class is one minus probability of the igneous disruption scenario class. Implementation of these probabilities in the TSPA is described in Sections 4.2 and 4.3.

The NRC acceptance criteria for the formation of scenario classes (NRC 2000 [149372], Section 4.2.4) address whether the scenario classes provide comprehensive coverage of all retained FEPs. The two scenario classes identified for TSPA-SR, nominal and igneous disruption, are broadly defined and mutually exclusive. All retained FEPs (both EFEPs and DFEPs) are contained in one or both of the scenario classes.

Step 5: Screening of Scenario Classes–In Step 4 above, 2 scenario classes were identified for screening: nominal and igneous disruption. These 2 scenario classes contain all retained FEPs. The relative probabilities of these 2 scenario classes are illustrated in Figure 2.1-4 and described in more detail in Section 4.3. The screening of scenario classes was performed to identify any scenario classes or scenarios that could be excluded from the TSPA on the basis of the same regulatory, probability, and consequence criteria defined in Step 3 for screening FEPs.

Scenario screening is used to identify scenarios that contain a combination of FEPs whose combined probability of occurrence (or consequence) is low enough to permit exclusion from the TSPA, even though the probability (or consequence) of the individual FEPs requires them to be retained. For a scenario class or scenario to be screened out, the combined low probability (or consequence) should not result from an inappropriately narrow scenario definition that artificially reduces the probability (or consequence) below the regulatory cutoff.

For TSPA-SR, detailed screening was performed on FEPs (as described in Step 3 above). The scenario classes and scenarios formed in Step 4 were composed of retained FEPs only. No additional exclusions were made during scenario screening. As was noted in Step 4, criticality was excluded from the TSPA at the FEP level rather than at the scenario level.

The nominal scenario class was implemented in TSPA-SR (Sections 3.1 to 3.9) by treating the retained expected FEPs through explicit incorporation in model components or through uncertainty included in the assignment of parameter values used in the model components. The igneous disruption scenario class (Section 3.10) and human intrusion scenario (Section 4.4) were implemented by treating the disruptive FEPs in similar fashion.

The NRC acceptance criteria for the screening of scenario classes (NRC 2000 [149372], Section 4.2.5) address the screening criteria and the appropriateness of applying the criteria to scenarios. For TSPA-SR, scenario screening criteria were evaluated, but all scenario classes and scenarios formed in Step 4 were retained.

2.1.2 Components of the Potential Yucca Mountain Repository System Evaluated in the Total System Performance Assessment

The potential Yucca Mountain repository system consists of the geologic setting and engineered barriers that, considered together, are aimed at reducing the exposure of humans to radioactive materials to acceptable levels. This section briefly describes the key aspects of the individual component models identified in Table 2.1-1 and Figures 2.1-5 to 2.1-8. Figure 2.1-5 depicts the general flow of information for the major scenario classes of the TSPA-SR and the components of these scenario classes. The scenario classes include the nominal (undisturbed) scenario class, the disruptive event (igneous/volcanic) scenario class, and the human intrusion scenario class.

- **Nominal Scenario Class**–Considers FEPs expected to occur during the time period of evaluation.

- **Disruptive Event Scenario Class**–Considers igneous disruption (i.e., volcanism) as an additional event that has a low probability of occurrence during the time period of evaluation. This scenario class includes two scenarios, igneous intrusion (indirect releases via groundwater) and volcanic disruption (direct releases).

- **Human Intrusion Scenario Class**–Considers a stylized event of human intrusion into the potential repository as defined in the governing regulations.

The nominal and disruptive scenario classes together contribute to the expected annual dose (Figure 2.1-5). Figures 2.1-6 to 2.1-8 show the individual flow-of-information wheels for the nominal scenario, the 2 disruptive scenarios, and the human intrusion scenario. These figures provide a visualization of how major information flows within each of the scenario classes and scenarios, and of the factors (or important submodels) for each of the components. Note that the nominal and human intrusion utilize essentially the same models and parameters, so these wheels look very similar.

The model components related to the first attribute of repository performance—limiting water contacting waste packages—include climate, infiltration, UZ flow, and seepage. Together, these components define the temporal and spatial distribution of water flow through the unsaturated tuffs above the water table at Yucca Mountain and the temporal and spatial distribution of water seeps into the repository drifts. There could be long-term (thousands to hundreds of thousands of years) climate variations. In addition, the thermal regime generated by the decay of the radioactive wastes can mobilize water over the first hundreds to thousands of years. For these reasons, the amount of water flowing in the rock and seeping into drifts is expected to vary with time.

The model components related to the second attribute of repository performance—prolonging waste package lifetime—include all of the above components plus EBS environments, drip shield degradation, and waste package degradation. Together, these components define the times when waste packages are expected to be breached. The thermal, hydrologic, and geochemical processes acting on the waste package surface are the most important environmental factors affecting the waste package containment time. As noted in Section 3.4, the mechanical degradation processes are currently estimated to be insignificant in affecting the containment time.

The environmental processes acting on the waste package surface as well as the timing and extent of waste package degradation are directly related to the selected design. Reviewing the key aspects of the site recommendation reference design as they relate to the expected behavior of the potential repository system is appropriate. Details of the reference design are described in Section 1.7 and are not repeated here. A schematic of the reference waste package design, including the types of waste forms to be emplaced in the potential Yucca Mountain repository, is depicted in Figure 2.1-9. Of particular relevance to performance are that the waste package reference design consists of 2 barrier metals—an inner structural liner consisting of 5 cm of stainless steel and an outer metallic barrier of 2 cm of corrosion-resistant high-nickel alloy (Alloy-22) (ASTM B 575-94 1994 [100497]) and the design and closure (post-weld heat treatment) of the lids of the waste package. For DOE-owned waste, the outer barrier is 2.5 cm thick.

The principal waste forms to be disposed of within these waste packages consist of the following:

- Commercial spent nuclear fuel derived from pressurized water reactors or boiling water reactors

- DSNFs including N-Reactor fuel from Hanford, Washington; research reactor fuel; and naval spent nuclear fuel

- High-level radioactive waste in the form of glass logs placed in stainless-steel canisters from Savannah River, South Carolina; West Valley, New York; Hanford, Washington; and the Idaho National Engineering and Environmental Laboratory, Idaho

- DOE-owned immobilized excess weapons-useable plutonium.

The waste packages are designed to contain up to 21 pressurized-water reactor assemblies, 44 boiling-water reactor assemblies, 5 glass logs and codisposal of DSNF fuel assemblies, and direct disposal of other canisterized DSNFs including naval spent nuclear fuel.

A schematic of the potential reference repository and EBS designs is depicted in Figure 2.1-10. Key aspects of the potential repository and EBS reference design that influence the long-term performance of the disposal system include the following:

- Areal thermal loading, which is determined by the waste package capacity, the spacing between waste packages and the spacing between emplacement drifts

- Size of the drifts

- Lining of the drifts for mechanical stability

- Characteristics of the engineered materials placed in the drifts to support the waste package (the waste package supports and inverts).

The model components related to the third attribute—limiting radionuclide mobilization and release from the EBS—include all of the above components plus seepage into the waste package, in-package chemistry, cladding degradation, colloid formation and stability, waste form degradation, and transport within the waste package. Together, these components lead to a determination of the spatial and temporal distribution of the mass of radioactive wastes released from the waste packages. Each component depends on the thermal, hydrologic, and geochemical conditions inside the waste package, which change with time.

The model components related to the fourth attribute of potential repository performance—slow transport of radionuclides away from the EBS—include all of the above components plus radionuclide transport through the EBS, the UZ, and the SZ; dilution from pumping; and radionuclide transport in the biosphere. Together, these components determine the spatial and temporal variation of radionuclide concentrations in groundwater. The groundwater concentration ultimately yields the mass of radionuclides that may be ingested or inhaled by individuals exposed to that groundwater, which in turn causes a level of radiological dose or risk associated with that potential exposure. Radionuclide transport may either be by advection (radionuclide movement that occurs with the bulk movement of the groundwater) or diffusion (radionuclide movement that occurs because of a concentration gradient). The concentration depends on both the mass release rate of the radionuclides as well as the volumetric flow of water along the different pathways in the different components. If the volumetric flow of water from the pumping well is greater than the volumetric flow in which the radionuclides are contained, then dilution of radionuclide concentrations can occur at the pumping well. The volume to be used in the TSPA-SR is based on water usage for the critical group, as defined by the proposed regulation.

Each of these key attributes and TSPA model components are used to evaluate the nominal performance of the potential Yucca Mountain repository system. These components describe the FEPs that are expected to occur throughout the period of interest (i.e., the expected FEPs). The FEPs that have a low (less than 1.0) probability of occurring over the period of interest (i.e., the disruptive FEPs), are considered in the disruptive event scenario class that is analyzed both separately and in combination with the nominal case. Human intrusion is analyzed separately (see Section 4.4) according to the stylized case defined in the proposed regulations.

2.1.3 Conceptual Description of Processes Relevant to an Evaluation of Postclosure Performance

The TSPA is an analysis of the long-term behavior of the repository system and the uncertainty in the analysis of that behavior. Before discussing how the analysis is performed (see Section 2.2), it is important to describe what is being analyzed. To describe what is being analyzed, it is necessary to describe the overall potential repository system and the components relevant to the evaluation of the repository system behavior.

The major components to be considered in the assessment of system performance and the relationship of those components to the repository safety strategy were presented in the previous sections. Described in this section are the key concepts of how the potential repository system is intended to work.

The basic principle of the potential Yucca Mountain repository safety strategy is to keep water away from the wastes. If water does contact the wastes, then the other principle of the safety strategy is to minimize the release rate of the radioactivity from the engineered barriers and reduce the concentration of any dissolved radionuclides as they migrate from the potential repository. The discussion that follows focuses on the small group of radionuclides that are mobile in the Yucca Mountain environment. Other radionuclides that either are very insoluble and/or highly retarded in the Yucca Mountain environment pose little risk to the environment.

Because the potential repository is approximately 300 m beneath the land surface and the wastes are solids (with minor gaseous constituents), the primary means for the radioactive constituents of the wastes to reach the biosphere, and ultimately humans, is along groundwater pathways. The wastes pose minimal risks to humans unless all of the following events occur:

- Waste forms are exposed to water.
- Radionuclides within these waste forms are dissolved in the water.
- Dissolved radionuclides are transported with the water.

- Radionuclide-containing water is discharged, either naturally or at a pumping well, from the aquifer.

- Humans or any part of the food chain uses this water.

If water is kept away from the wastes, the wastes pose little or no threat to humans.

The presence of water is of primary concern as it contacts the waste and as any dissolved radionuclides migrate within the groundwater to expose humans to the potential effects of radiation. One of the reasons for the primary issue being related to aqueous processes is that in the TSPA-SR, the performance measure of concern is dose to individuals. Although gaseous transport of some radionuclides (notably ^{129}I and ^{14}C) can occur, doses attributed to these release and transport mechanisms are expected to be insignificant. In following the water movement through Yucca Mountain, the major components and processes affecting the long-term isolation of radioactive wastes in the potential Yucca Mountain repository system are described. Also, this section depicts how the repository system is intended to work and provides a series of illustrations that picture the basic concepts that will be quantified in the TSPA.

In addition to tracking the movement of water through the repository system, the following discussion addresses a range of spatial and temporal scales. Being explicit in the definition of the scale being used is important because processes that might be explained at a spatial scale of kilometers must also be discussed at the scale of millimeters. Also, although time scales on the order of days and years are familiar concepts, it is sometimes difficult to extrapolate to the thousands or tens of thousands of years of importance in geologic systems. The discussion has been divided into six topics:

- Water movement in the unsaturated rocks above the potential repository

- Water and water vapor movement around the repository drifts

- Water movement within the EBS

- Water movement and radionuclide migration out of the EBS

- Water movement and radionuclide migration through the unsaturated tuffs below the potential repository

- Water movement and radionuclide migration through the SZ aquifers and biosphere.

Each of these areas is discussed below.

2.1.3.1 Water Movement in the Unsaturated Tuffs above the Potential Repository

Figure 2.1-11 illustrates the key concepts associated with water movement in the UZ at Yucca Mountain. Water at the repository horizon in the UZ at Yucca Mountain has as its source precipitation at the surface. This precipitation occurs as rainfall and snow and varies over time and space. The spatial variability is defined by precipitation being generally higher at higher elevations, such as along the crest of Yucca Mountain, and lower at lower elevations. The temporal variability is characterized by most of the precipitation occurring in the winter months or during brief summer thunderstorms, with the precipitation being higher during El Niño years. Because of long-term (thousands of years) climatic variations, the average precipitation in southern Nevada is expected to increase from current conditions. These long-term, transient precipitation changes may be affected by human-induced changes.

A significant fraction of the rainfall and snowmelt on the surface of Yucca Mountain either runs off into the washes that bisect the mountain, evaporates from the surface, or transpires from the native plants in the area. The remaining water continues downward through the soil horizon and eventually infiltrates into the rock. The net amount of total precipitation that infiltrates is called net infiltration. The net infiltration varies with space and time. The spatial variability is caused by variations in precipitation, soil conditions (permeability, thickness, and antecedent water content), geographic conditions (slope angle and slope direction), and vegetation conditions. The temporal variability is caused by the variability in precipitation.

The net infiltration of water moves downward through the UZ, driven primarily by gravity. In the UZ, this downward movement of water is called percolation flux to distinguish it from infiltration, or movement of water in the soil horizon. Some lateral diversion of water occurs as it moves downward from the soil horizon through the UZ. This lateral diversion is caused by the eastward dip of the geologic strata and the heterogeneities in the rock because of the different welded and nonwelded tuffaceous lithologic units between the surface and the potential repository. Although the water may be spatially and temporally distributed at depth, this distribution is generally a subdued reflection of the infiltration distribution at the surface because gravity drainage drives the groundwater flow system in the UZ (CRWMS M&O 2000 [145774]).

Water movement or flux in the unsaturated, fractured tuffs occurs in the matrix and the fractures of the rock. Generally, the welded tuff layers have more of the total flux within the fractures because the permeability of the matrix is low, while the nonwelded lithologic layers have more of the total flux within the matrix. Capillary forces tend to cause the water to move from the fractures, which are characterized as having a low suction, into the matrix, which has a high suction. This process is called matrix imbibition. The process is more prevalent in rock units with lower matrix saturation (e.g., the Tiva Canyon welded unit) and less significant when the matrix saturation is higher (e.g., the Topopah Spring welded units) or the fracture spacing is large (e.g., the Paintbrush non-welded unit) (CRWMS M&O 2000 [145774]).

2.1.3.2 Water and Water Vapor Movement around the Potential Repository Drifts

Figure 2.1-12 illustrates the key concepts associated with water movement around the repository drifts after waste emplacement at Yucca Mountain. Without heat-producing wastes in the drifts, the water in the unsaturated rocks around the repository drifts will tend to stay in the rocks and flow around the drifts rather than drip into the drifts. Water stays in the rocks because the rocks' capillary forces, including the fractures that contain most of the water flux, are greater than the gravitational forces required for causing a seep unless the fractures are almost fully saturated (see Section 3.2).

The characteristics of the rock around the potential repository openings may change with time. The fracture permeability could increase because of mechanical stress relaxation following the construction of the repository drifts and ultimately the collapse of the drifts. The fracture permeability may also change due to rock thermal expansion and mineral precipitation. The capillary suction of the fractures could either increase or decrease because of these same processes. However, these changes are expected to be within the range of natural variability existing before construction of the facility. The net amount of seepage and the fraction of the potential repository area in which seepage is expected to occur are important factors in the overall performance assessment because they determine the likelihood that individual waste packages will be contacted by seepage water (see Section 3.1).

Water seepage from the rock into the drifts will be affected during the operational phase of the facility by the ventilation of the potential repository. The ventilation will take moisture from the drifts and the rock in the form of water vapor.

Following waste emplacement, the heat generated by radionuclide decay will drive moisture in the rock away from the heat source, i.e., away from the spent nuclear fuel containers in drifts. This water will recondense in areas of lower temperature above, below, and between the hotter drifts. During the first few hundreds of years, there will be little or no seepage of liquid water into the drifts, because the water is generally being driven away. During this time, water in the drifts is in the form of water vapor or humid air. During the early periods, the RH in the drifts is reduced, but the RH eventually returns to close to 100 percent, as in ambient unventilated rock openings.

The distribution of liquid water and humid air within and around the repository drifts is variable in space and time. The spatial variability is caused by heterogeneity in the rock properties and variations in the ambient percolation flux. In addition, differences in the thermal output of different waste packages cause a range of thermal-hydrologic conditions in the potential repository. For example, cooler regions are expected along the edges of the potential repository and near low-thermal output waste packages. The temporal variability in water movement around the drifts is caused in the short-term by the thermal output of the wastes that eventually declines to minimal values (hundreds of years of drying and several thousand years of cooling and rewetting). In the long-term, the water movement is controlled by the climatic variability discussed in Section 2.1.3.1.

2.1.3.3 Water Movement within the Engineered Barrier System

Figure 2.1-13 illustrates the key concepts associated with water movement within the drifts and the contact of water with the waste package. Note that Figure 2.1-13 is only illustrative for waste packages experiencing seeps, which is expected to be a small fraction of the total number of waste packages. Water in aqueous or vapor form can cause slow degradation of the metallic waste package barrier. The dominant degradation mode of the outer Alloy-22 is by aqueous or humid air corrosion. At low relative humidities and in the absence of liquid water, the corrosion rate of Alloy-22 is generally low; however, at high relative humidities or in the presence of liquid water, this metal can corrode, albeit relatively slowly, exposing the inner structural liner composed of stainless steel.

The Alloy-22 layer generally degrades only in the presence of liquid water, i.e., when water drips directly on the waste package. Alloy-22 is generally immune to localized pitting and crevice corrosion and most failures will be by slow general corrosion, or by SCC.

The degradation rates of the stainless steel and high-nickel alloys are also affected by the temperature of the waste package surface, the chemistry of the water in contact with the waste package surface, the mechanical stress, and the degradation characteristics of the metals themselves. Because these environmental parameters are spatially variable and because the metal fabrication is variable, the waste package degradation is also expected to be variable in space and time. Not all of the waste packages are expected to be breached at the same time. In addition, the temporal variability in degradation rate implies that, once a single opening exists through the metallic waste package, it takes additional time before more openings penetrate through the waste package.

Until the same waste package has been sufficiently degraded to allow an opening to form through the two metallic barriers, there is no potential for water to come into contact with the wastes. During this period, the wastes are completely contained within the waste package. Once an opening exists, some of the seepage water falling on the waste package could enter the package.

2.1.3.4 Water Movement and Radionuclide Migration out of the Engineered Barrier System

Figure 2.1-14 illustrates the key concepts associated with water moving into the waste package and contacting the waste form. Also illustrated is migration through the EBS, of radionuclides that may exist as either dissolved species or adsorbed onto colloidal particles.

After the waste package has been breached, water may enter the waste package and contact the waste forms. For commercial spent nuclear fuel and many types of DOE-owned spent nuclear fuel, this water will first come into contact with the Zircaloy cladding around the spent nuclear fuel pellets. Zircaloy is a highly corrosion-resistant metal alloy; it is even more resistant to the effects of generalized or localized corrosion than Alloy-22. (Although Zircaloy has been considered as a candidate waste package material, the high cost of this alloy precludes its use in the site recommendation reference design.) Zircaloy will eventually degrade with time under several different mechanisms, but for a certain period it will prevent water from directly contacting the wastes. For high-level radioactive waste, a stainless-steel pour canister surrounds the waste glass. For much of the DOE-owned spent nuclear fuel, the wastes are contained within aluminum or Zircaloy cladding that is not fully intact, which in turn are planned to be placed in stainless steel or other metal alloy canisters. Aluminum and stainless steel are not as corrosion resistant as Zircaloy. When the material surrounding the actual waste form has degraded, the wastes are exposed to the environment inside the waste package and liquid water can contact a portion of the exposed waste. If water contacts the waste form, the radionuclides can dissolve in the water. Some radionuclides are highly soluble in water, while others are very insoluble in the water that is likely to contact the waste. Some radionuclides may attach to very small colloids that are mobile in the water.

When radionuclides are released from the solid waste form into the mobile liquid phase, they are available for transport. The transport mechanism depends on the distribution of water on the waste form surface and between the waste form surface and the outer edge of the degraded waste package. If water has dripped into the waste package, it is possible that advective transport of radionuclides to the edge of the waste package could occur. If the water has not dripped into the waste package, then a continuous, interconnected water film along which radionuclides may diffuse is required.

After radionuclides are transported through the degraded internal material of the waste package to the edge of the waste package, they may be transported through the degraded invert materials beneath the waste package. Radionuclides may be transported through the degraded invert by either moving water if there is seepage water, or diffusion through the pores of the invert materials. The radionuclides transported through the degraded invert are ultimately released to the tuff rock units to be transported in the UZ below the potential repository and ultimately to the SZ.

The rate at which radionuclides are released and transported from the potential repository depends on the following:

- Degradation rate of the engineered barriers
- Dissolution rate of the waste forms
- Form of the released radionuclides
- Solubility of the aqueous radionuclides
- Rate of water movement and volume of water that flows or diffuses through the engineered barriers.

2.1.3.5 Water Movement and Radionuclide Migration through the Unsaturated Tuffs below the Potential Repository

Figure 2.1-15 illustrates the key concepts associated with water movement in the unsaturated rocks beneath the potential repository and the migration of radionuclides in these rocks. After the dissolved or colloidal radionuclides are released into the unsaturated tuffs beneath the potential repository, they may be transported with the water to the water table. The rate at which these radionuclides are transported to the water table is a function of the following:

- Percolation flux in the unsaturated tuffs
- Distribution of the percolation flux between fractures and matrix
- Effective velocity of the groundwater within the fractured rocks
- Adsorption of radionuclides within the rock.

Because each of these characteristics of the natural environment is variable in space and time, radionuclide transport is also variable. Part of the temporal variability relates to long-term climatic changes that not only change the percolation flux through the system but also cause the water table beneath Yucca Mountain to rise (in the case of wetter climates) or fall (in the case of drier climates).

2.1.3.6 Water Movement and Radionuclide Migration through the Saturated Zone Aquifers and Biosphere

Radionuclides that are transported through the UZ are released to the saturated aquifers beneath the potential repository. Figure 2.1-16 illustrates the key concepts associated with water movement in the saturated aquifers beneath and downgradient from the Yucca Mountain site and the migration of radionuclides in these aquifers. Also illustrated are the pathways by which any dissolved radionuclides may come into contact with humans.

When the radionuclides reach the SZ, they will be transported laterally within the SZ. The general direction of groundwater flow in the SZ is to the southeast, and then possibly to the south and southwest. The concentration of the radionuclides in the aquifers at any point downgradient from the potential repository is a function of the following:

- Radionuclide concentrations in the water that enters the SZ
- Dispersion of these radionuclides as they are transported
- Adsorption of these radionuclides on the mineral surfaces along the flow path.

The time for radionuclides to reach any specified point downgradient from the potential repository, such as the 20-km point chosen for evaluating the system performance, depends primarily on the groundwater velocity and the retardation of radionuclides that may sorb on the mineral surfaces within the tuff or alluvial aquifers.

There is minimal risk associated with radionuclide releases as long as the concentration of radionuclides in water that is pumped from the aquifers downgradient from the potential repository is sufficiently low. Should radionuclides reach a location downgradient from the potential repository where water is pumped from the aquifer, the potential exists for radionuclides to come into contact with humans through biosphere pathways. The principal biosphere pathways to humans consist of the following:

- Direct consumption of water containing dissolved radionuclides

- Consumption of crops produced using water containing dissolved radionuclides

- Watering of livestock with contaminated water and/or feeding of livestock with contaminated crops, and the subsequent consumption of meat or milk

- Direct exposure to contaminated soil

- Inhalation of dust that may contain attached radionuclides.

The previous discussion outlined how the various components of the potential Yucca Mountain repository system fit together to describe how the system is intended to work. The general conceptual aspects of each key component and processes that affect the expected behavior of the repository system have been described. The next section (Section 2.2) describes the approach used to assemble the representations of the individual components into a description of the entire system. The details of each of the component models used in the TSPA and the scientific bases for these models are presented in Section 3 of this document, and in Section 6 of the *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]), as well as in the process model reports and analysis model reports.

2.2 METHODOLOGY

This section presents an overview of the method for mathematical and numerical modeling of each process and component introduced in Section 2.1, including their uncertainty, and the approach for combining them into an overall model and computer code. The overview includes discussions about information flow between the models (Section 2.2.1) and the computer code architecture that facilitates the information flow (Section 2.2.2). This section also provides a road map showing how to recouple the component models into one integral whole, as well as how to reassemble the analyzed pieces and pass information between them to develop reasonable assessments of overall system performance. The method for correctly coupling the component models to make robust predictions of repository behavior is composed of the basic activities outlined in Section 1.6.

2.2.1 Information Flow between Component Models

A stylized conceptualization of the TSPA-SR model hierarchy and information flow is shown in Figures 1.1-1 and 2.2-1. These figures indicate a continuum of information and models, from the most basic, detailed level to the level of the total system model. The data and associated conceptual and process-level models rest at the base of the pyramid (Figure 1.1-1). These process-level models may be simplified or abstracted ("abstraction" is used to connote the development of a simplified mathematical and/or numerical model that reproduces and bounds the results of an underlying detailed process model), if necessary, because of computational constraints or lack of information. Much of the modeling of the potential repository (and its components) is complex, uncertain, and variable, involving a variety of coupled processes (thermal-hydrologic-chemical and thermal-hydrologic-mechanical) operating in three spatial dimensions on a variety of different materials (e.g., fuel rods, waste packages, invert, and host rock) and changing over time. For these reasons, it is often necessary to make some simplifications to the detailed process-level models. The need for simplification is particularly evident in the TSPA, which has a significant component of probabilistic risk analysis.

The general approach of using probabilistic risk analysis is appropriate because of the inherent uncertainties in predicting physical behavior many thousands of years into the future in a geologic system with properties that can never be fully characterized deterministically. Because of the large number of uncertain parameters in the component TSPA models, probabilistic risk analysis involves a Monte Carlo method of multiple realizations of system behavior, which requires significant computational resources. For this reason, and because the lack of certain data makes some detailed models difficult to quantify, model abstractions are often employed. The abstracted performance assessment models may have a one-to-one correspondence with the detailed process-level models or may represent a combined subsystem model covering several aspects of the overall system. The performance assessment models form the components of the overall TSPA model at the top of the pyramid. Total system model simulations can then be performed in the computationally intensive probabilistic framework necessitated by a Monte Carlo approach to performance assessments.

For this model simplification process, there are two key factors in accurately representing the performance of the overall system. First, information and assumptions passed up the model pyramid must be consistent. For example, an infiltration flux used to generate liquid flow fields from the detailed process model for the UZ must be used in all subsequent analyses based on those particular flow fields. The same flux must be used when calculating seepage flux in the abstracted seepage subsystem model (Section 3.2) and when calculating thermal hydrologic response (temperature and relative humidity [RH]) in the near-field environment (Section 3.3). Second, the parameters that most affect performance in the detailed process models must be appropriately represented in the subsequent subsystem and total system models, including the appropriate uncertainty range of the parameters.

A key feature of the methodology is the approach utilized to pass uncertainty at one level to uncertainty at another level. Transfer of uncertainty must go in both directions, from bottom up and from top down. When analyzing uncertainty at the bottom levels (data, conceptual models, and process models), the analyses look at the effect of uncertain parameters on surrogate or subsystem performance measures, such as the amount of fracture flow in the UZ. The sensitivity of the surrogate measure to component model uncertainty is then used to decide whether to carry this uncertainty through to the total system analyses. However, sometimes important parameters at the subsystem level prove to be unimportant at the overall system level, and this information is then passed down the pyramid to indicate the relative unimportance of collecting more physical data about this parameter.

Traceability of data transfer among models and quality assurance of the data are very important aspects of the information flow process. The Process Model Reports and Analysis Model Reports, which support the TSPA results presented here, explicitly identify the sources and status of data, computer codes, and computer input and output files used in the Site Recommendation. Following prescribed procedures, the DOE is reviewing the data, assumptions, computer codes, and information used in the TSPA analyses to ensure the models are valid, defensible, and appropriate. To be fully qualified, there must be clear documentation that the TSPA models are supported by qualified data, and the numerical models and computer codes are documented and appropriately controlled.

Figure 2.2-1 is a more detailed, but still simplified, look at information flow among the component models: UZ flow (and seepage), thermal hydrology, engineered barrier system geochemistry, waste package and drip shield degradation, waste form degradation, EBS transport, UZ transport, SZ flow and transport, biosphere, and volcanism. It does not show all of the couplings among TSPA-SR component models but does illustrate major model connections, abstractions, and information feeds.

The information transfer between component models is activated in several ways. One approach is use of a "response surface," which means a multidimensional table of output from one model to be used as input in another model. When interpolating among points in the table, linearity is generally assumed. Usually a response surface has more than one independent variable (e.g., both time and percolation flux). However, in the usage in Figure 2.2-1, occasionally time is the only independent variable, and the data are provided directly "as is" to the next model, so no interpolation is required.

Figures 2.2-2a and 2.2-2b give a more detailed description of information flow in the TSPA-SR, showing the principal pieces of information passed between the various component models. Figure 2.2-2a shows the overall system, while Figure 2.2-2b shows the details of the engineered barrier system. These details of information flow are explained in greater depth in the discussion of the TSPA-SR code architecture in Section 2.2.2. The conceptual and experimental basis for this depiction of information flow is given in detail in Section 3. For example, the division of the repository horizon into five bins based on thermal hydrologic response and infiltration flux is discussed in Section 3.3; the division of the SZ water table into four regions, unrelated to the five repository bins (based on stratigraphy and other factors), is discussed in Section 3.8.

The decoupling of the physical-chemical processes into component models, shown in Figures 2.2-1, 2.2-2a and 2.2-2b, is facilitated by a natural division of the potential repository system into a series of sequentially linked spatial domains (e.g., the waste package, emplacement drift, host rock near the drift, UZ between the drift and the water table, SZ, and biosphere). This division works best from the standpoint of radionuclide transport, which is the primary consideration of the TSPA models. The TSPA-SR model architecture and information flow becomes, therefore, a sequential calculation in which each spatially based transport model may be run in succession, with output as "mass versus time" from an upstream spatial domain serving as the input of mass versus time for the spatial domain immediately downstream.

An additional complexity to this approach to transport is brought on by the inclusion of the disruptive event, volcanism, and its effects on the system. The systematic transport in the nominal system is disrupted or disconnected when volcanic events are included in the model. However, this is accommodated in the overall information flow by integrating the volcanism effects into the nominal model.

2.2.2 Code Architecture

The overall information flow, discussed in Section 2.2.1, forms the basis for the architecture of the TSPA-SR computer code. The executive driver program, or integrating shell, that links all the various component codes is GoldSim (Golder Associates 2000 [151202]). It is a probabilistic sampling program that ties all the component models, codes, and response surfaces together in a coherent structure that allows for consistent parameter sampling among the component models. The GoldSim program is used to conduct either single-realization runs of the entire system or multirealization runs of the system. The latter realizations yield a probability distribution of dose rate in the biosphere that shows uncertainty in dose rate based on uncertainty in all the component models.

Because of the need to conduct multiple realizations of the total system behavior, GoldSim is generally designed to model various components in a simplified fashion. However, the current version of GoldSim has some very useful features, such as cells and environments, that allow certain processes to be modeled in reasonable detail. The GoldSim program is also very flexible in representing various component processes in the total system model. The four ways that component models may be coupled into GoldSim, from most complex to least complex, include the following:

- External function calls to detailed process software codes (e.g., UZ transport software or waste package degradation software)
- Cells, which are basically equilibrium batch reactors that, linked in series, can provide a reasonably accurate description of transport through selected parts of the system (e.g., the engineered barrier system)
- Response surfaces, which take the form of multidimensional tables representing the results of modeling with detailed process models before running the TSPA code (e.g., thermal hydrology input)
- Functional or stochastic representations of a component model built directly into the GoldSim architecture.

The method used for each TSPA-SR component model is described briefly below and in greater detail in the corresponding parts of Section 3.

As described above for the third coupling method, much of the computational work that goes into the TSPA-SR model is done outside of GoldSim, before running the actual total system computations. For example, the UZ flow fields were computed using Transport of Unsaturated Groundwater and Heat (TOUGH2) (Pruess 1991 [100413]), a three-dimensional, finite-volume numerical simulator representing the entire UZ model domain (for the dual-permeability model). Other component models that were also run using computer codes outside of GoldSim include drift scale thermal hydrology (NUFT), the biosphere (GENII-S), in-drift and in-package chemistry (EQ3/6), and SZ radionuclide transport (FEHM). The results of these detailed process-level runs were provided as multidimensional tables that are read into GoldSim at run time. Examples of these multidimensional tables include (1) liquid flux and velocity fields for the UZ as a function of spatial position, time, and uncertain parameters such as infiltration flux and (2) temperature versus time.

Figure 2.2-3, in conjunction with Figure 2.2-1, provides a better understanding of the TSPA-SR code architecture (i.e., the actual computer codes used and the connections [information transfer] between codes). It includes both the codes run before the GoldSim program and those run in real time that are coupled to (external function calls), or within (cells and tables), the GoldSim program. Based on the schematic information transfer shown in Figure 2.2-3, some response surfaces generated by codes external to GoldSim only provide data to other codes external to GoldSim. Other response surfaces, such as liquid saturation, temperature, and seepage flux, will provide data directly into GoldSim as response surfaces that influence such things as waste form degradation rates. Not all couplings or all models are shown in Figure 2.2-3, (e.g., in-drift geochemical modeling is too complex to show all of its aspects in this figure) (Section 3.3).

Coupling of the various models is affected by the climate model, which impacts almost all the other models in one way or another, because it alters water flow throughout the system. The climate is assumed to shift in a series of step changes between three different climate states in the first 10,000 years: present-day climate, monsoon climate (about twice the precipitation of the dry climate), and glacial transition climate (colder than monsoon but similar precipitation). These climate shifts are implemented as a series of steady-state flow fields in the UZ and SZ (including changes in the water table elevation). Within the GoldSim program, these shifts require coordination among the coupled submodels because they must all simultaneously change to the appropriate climate state.

In general terms, the coding methods and couplings to be used for the major components are discussed below.

Mountain Scale, Unsaturated Zone Flow–This process is modeled directly with the three-dimensional, site-scale, UZ flow model (Section 3.2) developed by the YMP, using a volume-centered, integral-finite-difference, numerical flow simulator, called TOUGH2 (Pruess 1991 [100413]). Steady-state flow is assumed, and three-dimensional flow fields are generated for three different infiltration boundary conditions, three different climate states, and several values of rock properties. These "pregenerated" flow fields (i.e., developed externally and before the GoldSim simulations) are then placed in a library of files to be read by the finite element heat and mass (FEHM) code for UZ transport during the real time GoldSim simulations. Fracture and matrix liquid fluxes, along with liquid saturation, are passed to FEHM in these tables. To generate the library of flow fields, an inverse model, ITOUGH2 (Finsterle et al. 1996 [100393]) is used to calibrate the model-predicted ambient liquid saturations and other properties to measured liquid saturations and other properties in the matrix. This calibration is done when generating the flow fields for the three different infiltration conditions and the different fracture properties at present-day climate conditions. For future-climate conditions, flow fields are generated based on the present—day climate calibrations. Climate change is modeled within TSPA-SR UZ calculations by assuming a series of step changes in boundary conditions, meaning that different flow fields are provided at the appropriate time with the assumption of instantaneous pressure equilibrium. Based on the particular history of climate changes sampled by the TSPA model at the beginning of a given realization, the UZ flow field library is interrogated for a different flow field every time during the simulation that a step change is indicated. This change in a flow field is assumed to apply instantaneously to the transport model. The validity of this approach is discussed briefly in Section 3.7. The UZ flow fields are also provided to the TOUGH2 drift scale seepage models, to the SZ models, and to the engineered barrier system transport models. UZ hydrologic properties are passed to the drift scale, thermal hydrology model.

Seepage of Water into Emplacement Drifts (i.e., Drift Scale, Unsaturated Zone Flow)–This process is also modeled externally (Section 3.2) before the GoldSim simulations using TOUGH2 on a finely discretized grid around the drift and then abstracted for use in GoldSim. Simulations are conducted over a heterogeneous fracture permeability field (based on permeability measurements in the Exploratory Studies Facility) at a variety of percolation rates (from the mountain scale UZ flow model), and a variety of mean values and standard deviations for the fracture permeability distribution and the fracture "alpha" distribution (Section 3.2). These simulations become an uncertain response surface of seepage flux into the drift as a function of percolation flux and a response surface of the number of packages that are dripped on (by seeps) as a function of percolation flux. During the thermal pulse, the perturbed (increased) percolation flux is used as input to the response surface (increasing seepage), but no credit is taken for seepage reductions due to either evaporation of percolation flux entering the near-drift region or imbibition of percolation flux into dry pores in the same region; this is a conservative treatment of perturbed seepage.

Drift Scale, Unsaturated Zone Thermal Hydrology–This process is modeled with the finite-difference computer program NUFT (Nitao 1998 [100474]) in one, two, and three dimensions before the GoldSim simulations. The drift scale thermal-hydrology model uses a complicated set of embedded abstractions at different levels of spatial and process detail (e.g., conduction only versus conduction and convection), as described in Section 3.3. Outputs include:

- Waste package surface temperature and waste package surface RH for seven different package types within discrete environments. These values are provided to drip shield, waste package, and waste form models in GoldSim.

- Average waste form temperature and liquid saturation in the invert in each of the five repository level bins. Waste form surface temperature is actually assumed to be equal to the waste package surface temperature. These temperature and saturation values are provided to the waste form degradation and EBS transport models in the GoldSim program.

- Average drift wall temperature, RH, and liquid saturation in the invert in the potential repository. These values are provided to the EBS environment models. The outputs are in the form of response surfaces or multidimensional tables.

- Perturbed percolation flux above the drift. These values are used as inputs to the seepage response surface.

Engineered Barrier System Environment (i.e., Drift Scale Thermal Chemistry)–This process is modeled in the base case calculations outside of the GoldSim simulations by assuming a certain scenario for water flow through the drift and the types of materials the water contacts. Equilibrium batch-reaction calculations with EQ3/6 (Wolery 1992 [100836]; Wolery and Daveler 1992 [100097]) are performed at several places within the drift and then the output from one batch calculation is passed to the input of the next batch calculation at a different spatial location (Section 3.3). Output is a response surface of various chemical composition parameters. These values are provided to GoldSim directly as input tables for the waste form degradation and colloid models within GoldSim.

Drip Shield and Waste Package Degradation–This process is modeled within GoldSim using the WAPDEG computer code (CRWMS M&O 1998 [130755]), which includes corrosion-rate variability both on a given package and from package to package (Section 3.4). The code is linked to GoldSim and runs at the start of each realization to provide output in the form of several tables of the cumulative number of package failures per time, average patch area per package versus time, average crack area per package versus time, and average pit area per package versus time.

Cladding Degradation by Physical-Chemical Processes such as Creep Rupture–This process is modeled within GoldSim using functional relationships (and leads to a percentage value of failed cladding versus time exposed waste form area versus time [Section 3.5]). Other cladding degradation modes such as mechanical failure are also modeled within the GoldSim program. (CRWMS M&O 2000 [147210]) The major inputs to the cladding process model are measured characteristics (examples: oxide thickness, fission gas release) of commercial spent fuel which were collected and fit with first or second order equations. The input parameters for the abstraction are 1) peak waste package surface temperature, 2) water ingression rate into the waste package, and 3) temperature and chemical composition of the water inside the waste package.

Waste Form Degradation–This process is modeled as an equation within the GoldSim program using empirical degradation rate formulas developed from available data and experiments for the three different waste form types: CSNF, DSNF, and high-level radioactive waste (HWL) (Section 3.5). Output from the waste form degradation model is the mass of waste form exposed per time and the volume of water in contact with this waste form versus time, which is used directly in the GoldSim waste form cells. There are a variety of these waste form cells in the GoldSim program, corresponding to three different waste form types and several different seepage scenarios. The amount of inventory that can ultimately enter each waste form cell is a linear function of the number of packages emplaced in each inventory, seepage, and thermal hydrologic environment. There are 45 such environments, representing the product of 5 thermal-hydrologic regions, 3 inventory types, and 3 seepage environments. The entire waste inventory is composed of hundreds of different radionuclides. Of these hundreds, 39 were found to be present in sufficient quantity to warrant modeling in the near-field model components of the TSPA. Of these 39, only the 26 most important radionuclides—most important from the standpoint of delivering, or potentially delivering, the greatest dose rate at the biosphere location 20km downgradient of the repository—were tracked through all the system models. See Section 3.5 for details on the radionuclide inventory.

Engineered Barrier System Transport–This process is modeled directly within GoldSim at run time using the GoldSim cells algorithm. The modeling is based on an idealized representation (basically a linked series of equilibrium batch reactors) of drip shield, waste package, waste form, and invert, and how radionuclides move through them via diffusion and advection both as solutes and as colloids (Section 3.6). Output from EBS transport is radionuclide mass flux (for each of the modeled radionuclides) at each time step, passed during the GoldSim simulations to the directly coupled, three-dimensional, dual-permeability, FEHM particle tracker (Zyvoloski et al. 1995 [100528]) used for UZ transport. As shown in Figure 2.2-2a, the repository area is divided into five bins based on infiltration. The mass releases from these five source-term groups enter the grid blocks in FEHM that reside within the corresponding areas of the regions. The number of grid blocks receiving release is dependent on the number of packages failed. A key part of EBS transport is waste form or radionuclide mobilization, which is a direct function of both seepage flux and radionuclide solubility in the groundwater. Solubility for the various radionuclides is input directly into the GoldSim program in various forms (e.g., probability density functions, point values, and explicit functions). (See Section 3.5) Several types of colloid types are also modeled in the EBS transport component utilizing GoldSim functions. (See Sections 3.5 and 3.6).

Unsaturated Zone Transport–This process is modeled at run time using the directly coupled, three-dimensional, dual-permeability, finite-element code FEHM, which is accessed as an external function by the GoldSim program. Flow fields and property sets are accessed directly by FEHM from table files residing in the TSPA-SR controlled database. The UZ transport model is based on the UZ flow model and uses the same flow fields (generated by the TOUGH2 UZ flow code) and the same climate states. As with UZ flow, a dual-permeability model is assumed, and transport is modeled with the FEHM particle tracker in three dimensions. The FEHM particle tracker transports particles on the same dual-permeability TOUGH2 spatial grid as used in the flow model (using the same material properties, infiltration, and liquid saturation). When the climate shifts, a new TOUGH2 flow field is provided from the run-time file directory, and the particles are assumed to be instantly traveling with the new velocities. In addition, for multirealization runs, a matrix of uncertain UZ transport property values is created before simulation time by the GoldSim program and then accessed by FEHM during the simulations. The FEHM code steps through the uncertainty matrix row by row, where each row represents one realization of the uncertain UZ transport parameters, including K_ds (K_d is the measure of the partitioning of the mass of a given radionuclide sorbed or residing on the immobile rock phase to the mass dissolved in the aqueous phase) for each radionuclide, matrix diffusion coefficients, dispersivity, and K_c (K_c is the measure of the partitioning of the mass of a given radionuclide sorbed or residing on colloidal particles to the mass dissolved in the aqueous phase) values. Output from the FEHM code at each time step is mass flux from the fractures and matrix at the water table. The location of these output grid points is a vertical function of the climate state, increasing in elevation for wetter climates. The fracture and matrix mass fluxes from FEHM are combined appropriately for each of the 4 SZ capture zones in 4 GoldSim mixing cells and then fed to the SZ convolution integral SZ_CONVOLUTE at each GoldSim time step.

Saturated Zone Transport–This process is modeled using two models of SZ flow and transport (Section 3.8). A three-dimensional process level model (FEHM) is used to calculate, in detail, the transport of individual radionuclides important to dose. A one-dimensional flow tube model implemented in GoldSim is used to calculate the transport of daughter radionuclides (radionuclides that form by the decay of other radionuclides) of lesser importance. The models extend from 4 source regions at the bottom of the repository at the water table to the 20 km distance downgradient. The three-dimensional flow and transport simulations are done outside the GoldSim program for each of the selected radionuclides over 100 realizations of uncertain SZ model parameters. These uncertain parameters include effective porosity in the tuff and alluvium, K_d in the tuff and alluvium, colloid K_c, longitudinal dispersivity, fraction of flow path in the alluvium, and dilution factor (which mimics transverse dispersivity). The choice of 4 source regions is based on (1) examination of the releases from the UZ to the SZ showing roughly 4 areas of radionuclide input, and (2) minimization of source regions because each requires approximately 800 more breakthrough curves. Output from the FEHM stream tube simulations is concentration versus time at 20 km for a constant mass release rate source term. These breakthrough curves reside in files in the GoldSim run time directory and are accessed when needed by the SZ_CONVOLUTE external function (which convolves, or integrates, the real source term with the pregenerated unit breakthrough curves) called by the GoldSim program.

Biosphere Transport–This process is modeled within TSPA calculations using biosphere dose conversion factors that convert SZ radionuclide concentration to individual radiation dose rate. The biosphere dose-conversion factors are developed outside the GoldSim program using a computer program named GENII-S (Leigh et al. 1993 [100464]). The factors are then entered as table values in the GoldSim front-end menus. These factors are multiplied by the concentrations in the SZ stream tubes to compute individual doses, which are the end product of the calculations.

Disruptive Events–Igneous activity (indirect and direct volcanic effects) is modeled as a separate scenario. Seismic activity is modeled in the cladding model. Indirect volcanism is modeled within the TSPA model. This scenario utilizes many aspects of the nominal scenario and simply overlays an intrusive event, as characterized by its probability and physical properties (e.g., number of waste packages damaged by intrusion, extent of damage to waste packages, etc). After these effects are incorporated to the model, releases are handled as in the nominal scenario. Direct volcanic effects (i.e., radionuclides carried by ash plumes from volcanic eruptions) are modeled using the code ASHPLUME (LaPlante and Poor 1997 [101079]) that is directly coupled to the TSPA model at run time.

Human Intrusion–This scenario is analyzed separately, consistent with the scenario defined by the proposed regulations. The model is developed within the TSPA model utilizing inherent functions for release and transport of radionuclides assuming various conditions for the breach of the waste package. (See Section 4.4).

2.2.3 Testing of Integrated Total System Performance Assessment-Site Recommendation Model

This section presents the testing of the integrated Total System Performance Assessment-Site Recommendation Model. The performance assessment has been carried out using the code: GoldSim, which includes some process models, such as WAPDEG and FEHM. Also, several sub-models have been used to derive "abstractions" of other processes not directly simulated in GoldSim. An overview of the all computer codes used in this study, including their verification is presented in Section 3.0 of *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]). Further details of those codes is available from the relevant AMRs, which are also cited in the same document. Moreover, the performance of GoldSim, and its verification and validation in respect of each of the different physical processes modeled therein, such as, climate, infiltration, in-drift thermo-hydrology, UZ flow, waste package degradation, in-package chemistry, EBS transport, UZ transport, SZ flow and transport, biosphere doses, have been discussed in detail in Section 6.3 of the document cited. Also, Section 6.5 of the cited document does provide more details of the integrated model testing being presented here. In what follows, the strategy for the verification of the integrated model, *given* that the GoldSim model and its component models and the supporting models have been already verified, is discussed.

The verification and testing of the integrated TSPA-SR model has been conducted in two phases. The two phases are described below.

Phase 1:

In this phase, the computer model refers to a digital rendering of the conceptual model of the true physical system viz., the YMP site. The verification thus relates to checking that all aspects of the conceptual model are correctly implemented in the construction of the input for the simulation code GoldSim.

Phase 2:

Verification in this phase is directed to ensure that the simulation code GoldSim provides correct results for a given input (model). This verification was undertaken with a focus on the complexity in the different simulated processes related to the natural barrier system and the EBS, and also with a focus on architecture or the structure of the code. Figure 2.2-4 illustrates the 2 phases of this verification scheme. Verification of the code input and verification of the code output for a given input provides assurance that the performance assessment results are correct as modeled.

2.2.3.1 Phase-1: Verification of the Total System Performance Assessment-Site Recommendation Model

The verification of the TSPA-SR model (i.e., specifically the input model for GoldSim) consisted of assuring that the input construction is in complete accord with the conceptual models of the different processes as developed in a series of relevant and applicable analysis model reports. The conceptual models provided in various analysis model reports were converted into corresponding segments of the model input, which was then integrated to become the TSPA-SR model.

Translation of the conceptual model to the input model for the TSPA-SR model was accomplished and all components of the conceptual model have been reviewed to ensure that they are incorporated into the input. This activity of checking the input construction is arranged in a tabular form. This tabular form lists the different elements of the conceptual models and records their manner of incorporation in the input (as a data element, a function or an external dynamically linked library (DLL) routine). This tabular form utilizes an independent review process involving the author of each analysis model report, to ensure that the conceptual model was correctly translated into the corresponding segment of the input files. Figure 2.2-5 illustrates this procedure.

2.2.3.2 Phase-2: Verification of GoldSim

Phase 1 verification provides assurance that the TSPA-SR model is in full conformity with the conceptual model of the YMP site. Phase 2 verification ensures that the GoldSim model provides the correct output for a given input model embodying the full-scale complexity of the YMP site.

GoldSim has passed through a series of tests by its developers to demonstrate that it performs its numerical and logical operations correctly (Golder Associates 1998 [100449]). Nevertheless, considering the complexity of the processes simulated in the natural barrier system and EBS at the YMP, many of which are handled via external routines (e.g., WAPDEG, SEEP, FEHM), and considering the fact that some of these routines derive their input from the output of another preceding routine(s), the need to validate the code performance under the full-scale complexity of a realistic YMP model is warranted. Phase 2 verification addresses the verification of GoldSim from this perspective.

The Phase 2 verification consisted of three stages. Figure 2.2-6 explains stages 1 and 2, and Figure 2.2-7 explains stage 3.

Stage 1: Data Elements and Functions–GoldSim can compute some functions with data elements employing the user-prescribed functions, which can depend upon the intermediate output (e.g., waste form dissolution rate) of another process model. For example, the calculation of the pH values for seepage water entering the waste package and the solubilities of the radionuclides are all computed based on algebraic expressions, using the current values of some geochemical/thermal parameters, as computed by process models in the near-field environment. Such internal direct computations of some model (intermediate) outputs are verified by hand calculation at selected times, taking the output of the appropriate upstream process model as called for. Numerous examples of such verifications are provided in *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384], Section 6.3).

Stage 2: Dynamically Linked Library Routines–Besides the direct computations undertaken within GoldSim, major process simulations are performed via external routines (i.e., WAPDEG, FEHM). First, these routines are built as independent stand-alone codes and are verified. Second, they are incorporated into GoldSim as dynamically linked libraries. For the dynamically linked libraries, some data are transmitted from GoldSim through an argument list and other data are read from the data files, most of which are output from another process model within or outside GoldSim. The user also generates some data files read by a DLL through implementation of the applicable analysis model report for that process. The correctness of each of these types of inputs to a DLL was verified.

The verification of each individual process model, both as a stand-alone code and as a DLL in GoldSim, is being documented. (See for example, Verification Test Plan for WAPDEG [CRWMS M&O 1999 [149099]]).

Stage 3: Integrated Model Output Testing–When the full scale TSPA-SR model is implemented, where the inputs to a DLL are drawn from the outputs of another DLL, the final verification is done as follows.

The time-dependent inputs to a DLL are written to an output file of GoldSim, taking care to identify the DLL from where the data is printed. Those data are compared to the correct values of those input data, since they are known to be the outputs from another upstream dynamically linked library, which are also output from that dynamically linked library. This process ensures that the data transfers between the different process models are error-free. Since each DLL is validated individually under GoldSim, the verification of error-free data transfers between the models in GoldSim (when the integrated model is implemented) provides assurance that the output from GoldSim is correct, even for an input model which encompasses the full complexity of the conceptual model of the YMP site.

2.2.4 Treatment of Uncertainty in Total System Performance Assessment Analyses

2.2.4.1 Nature and Sources of Uncertainty

The assessment of long-term performance for the potential high-level radioactive waste repository at Yucca Mountain is a complex endeavor. It requires modeling various coupled hydrologic, geochemical, thermal, and/or mechanical processes taking place within the engineered and natural barriers over extended periods of time. In addition, the future evolution of the geologic and environmental conditions surrounding the disposal facility must also be considered, albeit in a somewhat stylized manner. Such integrated assessments of the future behavior of the disposal system are also complicated by uncertainties which arise due to incomplete understanding, limited information, and/or paucity of data. These uncertainties may be further categorized as follows:

- Scenario uncertainty
- Model uncertainty
- Parameter uncertainty and/or variability.

Scenario uncertainty stems from the fact that future evolution of geologic and environmental conditions surrounding the disposal facility, over tens of thousands of years, is inherently unpredictable. Scenarios of plausible future states of the system, and their likelihood of occurrence, must therefore be inferred from direct and/or indirect field evidence and incorporated into the performance assessment analyses. Examples of uncertain scenarios are (1) volcanic activity resulting in upward magma flow to the repository horizon and damage to waste containers and (2) change in climate from present-day conditions to a wetter, monsoon-type climatic regime.

Model uncertainty includes uncertainty in conceptual models and assumptions, uncertainty in mathematical descriptions of these conceptual models, as well as uncertainty in numerical implementations in computer codes. Because of incomplete understanding and characterization of FEPs, multiple plausible alternative conceptual models may be considered equally likely or defensible. This is often the major source of model uncertainty. Translation of a conceptual model into a mathematical model also results in uncertainties because of simplifications and approximations commonly employed to make the problem tractable. An example of model uncertainty is the representation of unsaturated flow at Yucca Mountain using the active fracture model (Liu et al. 1998 [105729]). Conceptually, the problem involves simplifying the characterization of water flow through a complex fractured rock mass using a simple dual-continuum fracture-matrix model. Additional uncertainty is introduced through the assumptions inherent in mathematical representations of fracture-matrix interaction and numerical solution of the governing equations, and calibration to field conditions using only a limited amount of data.

The parameters of the model used to predict the performance of the disposal system are also subject to uncertainty and/or variability. Uncertainty in model parameters arises because of imperfect knowledge or limited data and, in principle, can be reduced with additional measurements. For example, the solubility of neptunium in groundwater is not known with certainty but could be with enough additional measurements. Variability refers to the randomness or heterogeneity in physical and/or behavioral characteristics. It is an intrinsic property of the system and cannot be reduced by additional information. An example is the infiltration flux into the UZ at the surface of Yucca Mountain. Often, variability and uncertainty in a parameter are commingled because of imprecise knowledge. An example would be the seepage flux contacting waste packages. This flux varies from location to location within the repository horizon because of underlying heterogeneities in hydrogeologic properties. In addition, there is uncertainty about the value of flux at any given location because of limited characterization of the natural system.

This leads to a situation where the inputs of the TSPA model (i.e., scenarios, mathematical and conceptual models, and parameters) are uncertain and/or variable, which will therefore result in the output of the model being uncertain as well. As described in the following sections, a probabilistic framework has been adopted in TSPA-SR for translating uncertainties in model inputs to corresponding uncertainties in model predictions. This approach is also consistent with the regulatory standards proposed by the NRC and the EPA.

2.2.4.2 Regulatory Drivers

The NRC currently is in the process of developing the standard that will apply to the disposal of high-level radioactive wastes in the potential repository at Yucca Mountain (proposed 10 CFR Part 63 [64 FR 8640] [101680]). In the Supplementary Information published with the rule, the NRC has stipulated the application of a probabilistic framework for TSPA:

> Demonstration of compliance with the postclosure performance objective specified at § 63.113(b) requires a performance assessment that quantitatively estimates the expected annual dose, over the compliance period and weighted by probability of occurrence, to the average member of the critical group. Performance assessment is a systematic analysis of what can happen at the repository after permanent closure, how likely it is to happen, and what can result, in terms of dose to the average member of the critical group. Taking into account, as appropriate, the uncertainties associated with data, methods, and assumptions used to quantify repository performance, the performance assessment is expected to provide a quantitative evaluation of the overall system's ability to achieve the performance objective (64 FR 8640 [101680]).

Note that the NRC not only anticipates that there will be significant uncertainties (proposed 10 CFR 63.101), but the NRC also requires the TSPA take into account uncertainties in characterizing and modeling the barriers (proposed 10 CFR 63.114 [64 FR 8640 [101680]]). Furthermore, proposed 10 CFR 63.113(b) (64 FR 8640 [101680]) requires a demonstration of compliance by calculating an expected annual dose, defined as follows:

> The expected annual dose is the expected value of the annual dose considering the probability of the occurrence of the events and the uncertainty, or variability, in parameter values used to describe the behavior of the geologic repository (the expected annual dose is calculated by accumulating the dose estimates for each year, where the dose estimates are weighted by the probability of the events and the parameters leading to the dose estimate) (64 FR 8640 [101680]).

The EPA has recently proposed public health and safety standards in proposed 40 CFR Part 197 (64 FR 46976 [105065]), with which the potential repository at Yucca Mountain must comply. The EPA has also specified the application of a probabilistic framework where uncertainties associated with scenarios, models, and parameters are explicitly incorporated into the performance assessments for demonstration of compliance. The regulation specified by the NRC in proposed 10 CFR Part 63 (64 FR 8640 [101680]) is intended to implement EPA's standards and must ultimately be consistent with the EPA requirements.

2.2.4.3 Quantified and Unquantified Uncertainties

In the process models and the TSPA model, some uncertainties are quantified and others are unquantified. Quantified uncertainties are those for which a detailed, unbiased quantitative description of uncertainty (e.g., probability distributions) have been developed from available data. An unbiased estimate is one that neither deliberately overstates nor deliberately understates the uncertain quantity being estimated. In some cases, unbiased quantitative estimates of uncertainty are not feasible, for example, because of limited availability of data or process complexity. Unquantified uncertainties include alternative models, alternative hypotheses, assumptions and/or single point parameter values being used to represent an uncertain property. Unquantified uncertainty may also exist when model inputs are treated probabilistically, if the range of the inputs is chosen to be a conservative bound on the range of all possible values. Uncertainty that results from the plausibility of alternative conceptual models is often unquantified as well.

The computational framework for the TSPA-SR is the primary vehicle by which uncertainties are modeled and their impacts estimated and communicated. It is designed with sufficient flexibility that a range of parameters and models can be used to describe each component of the system. That flexibility also makes it possible to conduct extensive sensitivity and uncertainty analyses to estimate the impact and importance of modeled uncertainties on overall dose estimates. The strategy adopted in TSPA-SR has been to use unbiased and reasonable descriptions of the models and parameters, and their associated uncertainties, in the process models and abstractions when possible. When the processes of interest are too complex to model defensibly, and/or there is insufficient data to defend a specific definition of a model or parameter, a conservative bounded approach is used.

Defensible, in this sense, refers to models and data that can be defended in a regulatory environment on the basis of information available for use in the SR at the time it is submitted. The word conservative implies that models and parameters are chosen such that the ultimate dose estimates produced by the model will be deliberately overestimated. The use of such conservative estimates can sometimes eliminate the need to explicitly consider and model overly complex, sometimes intractable, uncertainties. This approach was chosen because it results in some modeling simplifications, and it is believed to result in dose estimates that can be defended in a regulatory setting as being conservative. In addition, the conservative approach is supported by peer review and regulatory perspectives, and there is precedence for the approach in some other international performance assessment programs.

Note that such a hybrid approach, consisting of a mixture of credible probability distributions and conservative single-point estimates and/or models, tends to lead to a projected outcome that is conservative – although the degree of conservatism implied by the results cannot be readily ascertained. Also note that the significance of unquantifed uncertainties can be masked by the impact of those quantities/models included in the model as quantified uncertainties.

2.2.4.4 Probabilistic Framework

The probabilistic framework used in TSPA calculations is a well-established methodology for incorporating the effects of quantified uncertainties in scenarios, conceptual models, and/or parameters. It has been extensively used in probabilistic risk analyses for evaluating the safety of nuclear reactors and power plants (Rechard 1999 [145383]). Several probabilistic performance assessments have also been carried out within the U.S. radioactive waste disposal program. These include a series of performance assessment studies for the disposal of transuranic waste at the Waste Isolation Pilot Plant (Helton et al. 1998 [100951]), as well as a series of calculations performed for the disposal of high-level radioactive waste at Yucca Mountain by the DOE (Barnard et al. 1992 [100309]; Wilson et al. 1994 [100191]; CRWMS M&O 1994 [100111]; CRWMS M&O 1996 [100962]; CRWMS M&O 1998 [108000]) and the NRC (Codell et al. 1992 [103714]; Wescott et al. 1995 [100476]).

Monte Carlo simulation, the most commonly employed technique for implementing the probabilistic framework in engineering and scientific analyses, is a numerical method for solving problems by random sampling (Morgan et al. 1990 [149538]). As shown in Figure 2.2-8, this method allows a full mapping of the uncertainty in model parameters (inputs) and future system states (scenarios), expressed as probability distributions, into the corresponding uncertainty in model predictions (output), which is also expressed in terms of a probability distribution. Uncertainty in the model outcome is quantified via multiple model calculations using parameter values and future states drawn randomly from prescribed probability distributions. Monte Carlo simulation is also known as the method of statistical trials because it uses multiple realizations of the inputs to compute a probabilistic outcome.

In some situations, the uncertainty in model inputs (e.g., processes, parameters) cannot be quantified using statistical approaches, either because of insufficient information or due to significant complexity. In TSPA-SR, such unquantified uncertainties have been generally represented in a bounded or a conservative manner, as summarized in Appendix F.

The probabilistic modeling approach is computationally burdensome because it requires several hundred model calculations for each scenario of interest. However, it also provides important information not available from a deterministic "best-guess" or "worst-case" calculation. The benefits of probabilistic modeling include (1) obtaining the full range of possible outcomes (and the likelihood of each outcome) to quantify predictive uncertainty and (2) analyzing the relationship between the uncertain inputs and the uncertain outputs to provide insight into the most important parameters.

2.2.4.5 Propagation of Uncertainty

A Monte Carlo analysis of the TSPA model involves the following four steps:

1. Select imprecisely known model input parameters to be sampled
2. Construct probability distribution functions for each of these parameters
3. Generate a sample set by selecting a parameter value from each distribution
4. Calculate the model outcome for each sample set and aggregate results for all samples (equally likely parameter sets).

A brief explanation of each of these steps is described in the following paragraphs. Note that in the TSPA-SR methodology, the Monte Carlo approach is applied to each scenario separately, and the results are combined based on the probability of each scenario.

Selecting Imprecisely Known Model Input Parameters To Be Sampled–The TSPA-SR model consists of approximately 1,000 parameters, many of which are uncertain and/or variable. A determination as to which of these have a significant range of uncertainty or variability, and thus need to be statistically sampled during model calculations, is made during the development of individual process models and/or abstractions thereof. Further discussion on the selection of these parameters can be found in the TSPA-SR model analysis model report (CRWMS M&O 2000 [148384]).

Constructing Probability Distribution Functions for Each Parameter–The probabilistic framework employed in Monte Carlo simulations requires that the uncertainty in model inputs be quantified using probability distributions. As noted earlier, not all uncertain parameters have been treated probabilistically in TSPA-SR. Some parameters have been chosen to be represented by conservative or bounding single-point values rather than probability distributions. Examples of such representations can be found in the descriptions of various TSPA component models in Sections 3.2 to 3.10, as well as in Appendix F. As in previous TSPAs (e.g., CRWMS M&O 1998 [108004]), a variety of sources have been used for generating information related to the distribution of stochastic inputs:

- Actual measurements (e.g., porosity of various hydrogeologic units)
- Expert elicitation (e.g., dilution factors in the SZ)
- Process model output (e.g., fraction of waste packages exposed to seeps).

Per NRC and EPA guidance, the construction of probability distributions has focused on the full range of defensible and reasonable parameter distributions that can be justified on the basis of available information and/or expert elicitation. These distributions are specified either as empirical distribution functions (i.e., individual values and their likelihood), or as the coefficients of parametric distributions fit to data (e.g., mean and standard deviation of a normal distribution fit to porosity data).

Generating a Sample Set by Selecting a Parameter Value from Each Distribution–The next step in the Monte Carlo process requires generating a number of equally likely input data sets, which consist of parameter values randomly sampled from the prescribed range and distributions. An improved form of random sampling is the Latin hypercube sampling procedure, where the range of each parameter is divided into several intervals of equal probability and a value is selected at random from each interval (Helton 1993 [100452]). Latin hypercube sampling, which is employed in TSPA-SR, helps achieve a more uniform coverage of the uncertain parameter range as compared to purely random sampling. The issue of interdependence or statistical correlation between parameters is also important from the perspective of maintaining the necessary dependence between random variable pairs. The sampling algorithm used in TSPA-SR ensures that any desired correlation between input parameters is retained.

Calculating Outcomes for the Sample Set and Aggregating Results for All Samples–In this step of the Monte Carlo methodology, the model describing the behavior of the system for the scenario of interest is evaluated for each of the randomly generated parameter sets. This is a simple operation consisting of multiple model calls, where the outcome (i.e., annual dose as a function of time) is computed for each sampled parameter set. Once all of the required model runs have been completed, the overall uncertainty in the predicted outcome can be characterized by (1) summary statistics such as the mean and median and (2) the cumulative probability distribution.

The NRC regulations in proposed 10 CFR 63.113(b) (64 FR 8640 [101680]) require that the mean annual dose history be computed for each scenario (nominal as well as disruptive) and weighted by the probability of that scenario to determine the overall expected annual dose history over the 10,000-year compliance period. The EPA regulations in proposed 40 CFR 197.13 (64 FR 46976 [105065]) require compliance demonstration using the mean or median dose history, whichever is higher. The regulations are thus focusing on the statistical average result, although further analyses of the full range of model outcomes will be necessary for developing reasonable assurance arguments for the NRC and reasonable expectation arguments for the EPA. The cumulative probability distribution is also useful for unraveling the relationship between input and output variables implicit in the TSPA model and for determining the relative importance of various uncertain inputs (Section 2.2.5).

As noted earlier, the procedure outlined above for translating input uncertainties into output uncertainties reflects the effects of the quantified uncertainties. Because unquantified uncertainties are represented via conservative/bounding values, their impact is believed to be restricted to over-estimating system performance in a pessimistic direction (i.e., increasing expected value of individual dose). It is also important to point out the uncertainty associated with developing models of credible features, events, and processes, based on scientific observations and/or inferences, is not explicitly incorporated in this methodology. The decision was generally made to focus on the most reasonably defensible model representation, but to err on the side of conservatism so as not to under-predict the possible consequences or risks.

2.2.4.6 Presentation of Uncertainty Analysis Results

For any given scenario (nominal or disruptive), the Monte Carlo methodology requires the TSPA-SR computer model to be evaluated for each of the equiprobable parameter sets sampled from their prescribed distributions. Thus, each realization results in a total dose history (i.e., table of dose rate as a function of time). The aggregation of all dose histories produces a picture of the overall uncertainty in predicted performance. Such information may be graphically presented in several ways:

- A graph of dose rate versus time showing results from all realizations. This is the so-called "horse-tail plot," which provides an indication of the overall spread in model results given the uncertainties in the inputs.

- Superimposing the mean dose rate history on the horse-tail plot. This is the arithmetic average dose calculated for each point in time. It is also the benchmark quantity for comparing the performance of the disposal system against regulatory standards.

- Superimposing the median dose rate history on the horse-tail plot. For each point in time, the median dose is the value above which lie 50 percent of the results and below which lie 50 percent of the results. The median dose rate history is a potential benchmark quantity per the EPA regulations.

- Superimposing the 5th and 95th percentile dose rate histories on the horse-tail plot. For each point in time, the 5th (95th) percentile dose is that value below which lie 5 percent (95 percent) of the results. These two values provide a reasonable indication as to the spread in computed model outcomes.

Figure 2.2-9 presents an example of such a composite graph, showing the dose rate history for all realizations, along with the median, the mean, and the 5th and 95th percentile dose rate histories. Note that this figure is for demonstration purposes only, and does not contain any actual results.

2.2.5 Sensitivity Analyses

2.2.5.1 Objectives of Sensitivity Analyses

The TSPA-SR model represents the behavior of a complex system with hundreds of parameters. Many of the parameters are uncertain and/or variable, and their interaction with one another can also be complex and/or highly nonlinear. It is difficult to obtain an understanding exactly how the model works and what the critical uncertainties and sensitivities are from a simple evaluation of model results. To this end, sensitivity analysis provides a useful and structured framework for unraveling the results of probabilistic performance assessments by examining the sensitivity of the TSPA-SR model results (and their uncertainties) to the uncertainties and assumptions in model inputs. Specifically, the goal of sensitivity analysis is to answer questions such as:

- Which uncertain variables have the greatest impact on the overall uncertainty (spread) in probabilistic model outcomes?

- How do different variables affect the model outcome when their values are varied over a range?

- How robust are the probabilistic model results to underlying assumptions regarding bounding/conservative values and/or conceptual models?

TSPA-SR uses uncertainty importance analysis, explanatory "one-off" analysis, and robustness analysis, respectively, to answer these questions. Details of each of these methods are described in the following sections.

2.2.5.2 Uncertainty Importance Analysis

The objective of uncertainty importance analysis is to identify those input parameters that have the greatest influence on the spread (uncertainty) of the probabilistic model results. This is accomplished qualitatively using scatter plots, and quantitatively using statistical methods such as correlation-regression analysis and classification and regression tree analysis. The analyses are carried out using results from the probabilistic calculations at a fixed point in time, with the sampled inputs corresponding to each of the realizations being treated as independent variables and the computed outputs being treated as dependent variables. Note that the outputs can either be total system-level performance measures, such as annual dose rate to a receptor, or they can be subsystem-level performance measures, such as cumulative radionuclide mass flux at the water table.

Scatter Plot Analysis–A scatter plot is a simple graphical tool for determining the strength of the relationship between an uncertain input parameter and the calculated output variable. Sampled values of the input parameter are plotted against the corresponding computed outcomes, after transforming the actual numerical values into ranks (i.e., the lowest value has rank 1, the next highest value has rank 2, and so on). If little or no relationship exists between an independent variable and the model outcome, the scatter plot will resemble a random distribution of points. However, if a significant relationship does exist, the plotted points will cluster and exhibit a recognizable form—either as an upward-trending cloud (meaning that the input-output relationship is positive) or as a downward-trending cloud (meaning that the input-output relationship is negative). Scatter plots also help reveal threshold phenomena and nonlinearities (or lack thereof) in the input-output relationship. Figure 2.2-10 shows some example scatter plots between two random variables. The top panel shows an upward trending relationship, the middle panel shows no apparent relationship, and the bottom panel shows a downward trending relationship until the threshold for an upward trending relationship is encountered.

Correlation and Regression Analysis–Scatter plots are valuable for identifying a relationship between an input variable and an output variable, but they do not quantify the intensity of that relationship. In performance assessment studies, multiple linear regression modeling is commonly used for this purpose (Helton 1993 [100452]). Using regression models, it is possible to identify input variables that contribute the most to the calculated uncertainty (variance) in the performance measure. The primary technique for regression modeling is stepwise linear regression using rank transformations of the input and output values. In the stepwise approach, a sequence of regression models is constructed starting with a single selected input parameter (usually the parameter that explains the largest amount of variance in the output), and including one additional input variable at each successive step (usually the parameter that explains the next-largest amount of variance) until all of the input variables that explain statistically significant amounts of variance in the output have been included in the model. This approach avoids having to treat all of the independent uncertain variables simultaneously in a single model.

Two indicators are used to rank the input variables: partial rank correlation coefficient and R^2-loss (where R^2 denotes the coefficient of determination for the regression model). Both of these indicators are calculated during stepwise regression modeling. The partial rank correlation coefficient for a particular input variable measures the correlation between the output and the selected input variable, after the linear influences of the other variables in regression have been eliminated (Helton 1993 [100452]). The second importance indicator used, R^2-loss, represents the loss in R^2 of the current n-variable regression model, if the variable of concern is dropped from the regression model (RamaRao et al. 1998 [100487]). A large value of R^2-loss (i.e., a large decrease in explanatory power) indicates that the removed variable explained a large proportion of the variance in the output and, therefore, the variable is an important component of the model. In TSPA-SR, the R^2-loss value is divided by the regression R^2 in order to facilitate the comparison of importance ranking from different times and/or for different simulations. This normalized metric, defined as the uncertainty importance factor, is essentially equivalent to the fraction of total variance explained by the variable of interest.

Classification and Regression Trees Analysis–Linear regression is useful for analyzing entire spectra of output data. However, analyzing small categories of output data may require a more specialized approach. Classification and regression tree analysis is a categorical method for determining what variables or interactions of variables drive output into particular categories (Breiman et al. 1998 [151294]). For example, classification and regression tree can be used to analyze the extreme values in a set of output data. Those realizations that yield the highest and lowest outcomes are grouped into high and low categories. The classification and regression tree analysis will then provide insight into what variable or variables are most important in determining whether outputs fall in one or the other category.

A binary decision tree is at the heart of the classification and regression tree analysis. The decision tree is generated by recursively finding the variable splits that best separate the output into groups where a single category dominates. The domination of a single category resulting from a split is called the purity of that split. For each successive fork of the binary decision tree, the classification and regression tree algorithm searches through the variables one by one to find the purest split within each variable. The splits are then compared among all the variables to find the best split for that fork. The process is repeated until all groups contain a single category. In general, the variables that are chosen by the algorithm for the first several splits are most important, with less important variables involved in the splitting near the terminal end of the tree.

The Figure 2.2-11 depicts a binary decision tree generated from a classification and regression tree analysis. The realizations which yielded the 30 highest and 30 lowest (including zero) outputs were grouped into high and low categories. Starting at the left side of the figure, the first split is based on the variable X1. If (X1 > 1) then the upper branch is followed, whereas if (X1 < 1) then the lower branch is followed. This split yields two groups: the upper branch contains 29 high values and 8 low values, while the lower branch contains 20 low values. The lower branch is pure so no additional splitting is required. The upper branch continues to a second split based on X2. This split divides the 29 high and 8 remaining low values into groups. All groups are now pure so the tree is terminated. Note that for simplicity, some branches of low importance may be left off of the tree. In this example, 29 of 30 high outputs and 28 of 30 low outputs are represented. The classification and regression tree analysis in this example has shown that the input values of variables X1 and X2 determine whether high or low outputs will result.

2.2.5.3 Probabilistic One-Off Analysis

A "one-off" analysis is a variation on some reference case calculation where the parameter of interest is modified from its original value while all other parameters are kept unchanged. In a probabilistic one-off analysis, as implemented in TSPA-SR, the probability distribution for the parameter of interest is replaced with its 5th or 95th percentile value (whichever yields the more conservative model outcome). Thus, the parameter of interest is given a single value while all other parameters are characterized using their full probability distributions.

The motivation for carrying out such an analysis is twofold. First, it allows an exploration of the sensitivity of model performance to extreme (but realistic) values in model parameters. Second, the analysis provides an indication of what can happen when the system is stressed to the extent that the parameter of interest is assigned a value which has only a relatively low likelihood of occurrence. This analysis is not necessarily intended to show how the reference system behaves, rather it suggests how the reference system is resilient when its parameters take on pessimistic values.

With respect to the actual implementation of the methodology, the first step is to screen for candidate parameters based on the results of uncertainty importance analysis. The objective here is to identify those parameters important at the system level (affecting receptor dose) as well as the subsystem level (affecting waste package failure, EBS release, UZ release, etc.). The next step is to pick the 5th or the 95th percentile value at which these parameters would be fixed during the one-off calculations. The probabilistic calculations with this modified parameter set are then carried out, and the expected dose (or other outcomes of interest) compared against the corresponding result from the reference case.

2.2.5.4 Robustness Analysis

The TSPA-SR model includes parameters which are treated as constants, parameters described via probability distributions to represent inherent uncertainty and/or variability, as well as imprecisely known parameters represented with conservative and/or bounding values. Because of this mixture of representations, it has been pointed out that care should be taken in interpreting the results of statistical sensitivity or uncertainty importance analyses (Budnitz et al. 1999, [102726]).

In order to further analyze results from models with such heterogeneous sources of information about uncertainty (i.e., probability distributions versus bounding values), TSPA-SR uses a modified form of the range/confidence estimate approach proposed by Richards and Rowe (1999 [148939]). Additional probabilistic analyses are used to explore the robustness of the reference probabilistic model results to the underlying assumptions of imprecise parameter representations. The probabilistic one-off analyses described previously are restricted to the 5th and 95th percentiles of the distributions used in the reference probabilistic case. In robustness analyses, the shape and the range of the distribution itself can be changed based on new information, alternative points of view and/or what-if scenario assumptions. Another form of robustness analysis focuses on parameters described with bounding/conservative values. Utilizing more realistic values in a manner similar to the range/confidence estimating protocol suggested by Richards and Rowe (1999 [148939]) provides an indication of the degree of conservatism in the original results. Uncertainty importance analyses for the scenario with these modified parameters also shows the sensitivity of the importance rankings to underlying uncertainties in parameter representations.

2.2.5.5 Sensitivity Analyses and Reasonable Assurance

In proposed 10 CFR Part 63 (64 FR 8640 [101680]), the NRC recognizes that complete assurance of compliance with regulatory standards cannot be obtained based solely on the results of probabilistic performance assessments because of the uncertainties inherent in the understanding of the evolution of the geologic setting, biosphere, and EBS. Even though the TSPA-SR model incorporates the best state of current knowledge with respect to the uncertainties in its component models and parameters, residual uncertainties are likely to remain. The NRC therefore requires a demonstration of reasonable assurance, making allowance for the time period, hazards, and uncertainties involved, that the predicted outcome will satisfy the prescribed performance objectives. The suite of sensitivity analyses used in TSPA-SR support the development of reasonable assurance arguments in the following manner:

- Identifying the key uncertain parameters so that more effort can be directed at minimizing these uncertainties, if possible

- Examining what happens when the system is stressed via unfavorable parameter values and/or conceptual models to obtain a better sense of the range/confidence of performance predictions

- Testing the robustness of predicted model outcomes to underlying assumptions about model and parameter structure.

By addressing the importance of known uncertainties with respect to conclusions and the issue of confidence in the TSPA-SR model results, the sensitivity analyses provide additional lines of evidence toward building reasonable assurance.

2.2.6 Control of Information in TSPA

The TSPA-SR model utilizes information from a large number of sources, including AMR's, literature data, and information housed within the Technical Data Management System. In all, there are over 6,500 parameter values within the TSPA model, plus over 20 data tables attached to the model file, as well as the 4 external process models (i.e., ASHPLUME, WAPDEG, FEHM particle tracker, SZ_CONVOLUTE) and 3 software routines (i.e., Seep, Soil, GVP) attached to the model.

The receipt and use of this information is controlled procedurally primarily by *Transmittal of Input*, AP-3.14Q [152629]; *Managing Technical Product Inputs*, AP-3.15Q [153184]; *Submittal and Incorporation of Data to the Technical Data Management System*, AP-SIII.3Q [149901]; *Software Management*, AP-SI.1Q [153201]; and *Analyses and Models*, AP-3.10Q [152363].

These procedures provide the protocol for transmittal and utilization of the information in a controlled, traceable fashion.

The status of the data, software or models in this technical product is presented in several ways. The software status is shown in Table 2.2-1. All qualified software used in the TSPA-SR analyses were obtained from Configuration Management and used within the range of validation. The unqualified software used in the TSPA-SR analyses are being controlled in accordance with AP-SI-1Q [153201], and AP-3.15Q [153184], as indicated in the table and in the associated DIRS. The major software inputs are documented in the TSPA-SR model document. Other inputs (as well as outputs) are documented in Appendices E and G.

Changes to this report may be required as confirmation activities associated with unresolved TBX's and Urn inputs are completed. Software qualification may also lead to changes in the analyses if additional simulations are required. The input status of the data and models utilized in this technical product are indicated for the references in the DIRS. Status of inputs need to be indicated in the DIRS.

Figure 2.2-12 schematically shows the major components of data, codes or software, and models that must be controlled. Data are developed or acquired, submitted to the Technical Data Management System and given a data tracking number that provides traceability to the specific information utilized. The data tracking number identifies the source, type of data, and who can be contacted to find out more about the data. Software or codes required to run the models are also controlled, both during initial development and after maturity. The Software Configuration Management Organization is responsible for centralized control of the software. The qualification procedure requires testing and documentation of the software in a very thorough manner. This process provides a thorough check that the software is calculating what it was designed to calculate as long as it is used within its design specifications. Models, and submodels, are also developed and controlled in a manner that provides suitable documentation and review of the assumptions (inputs and outputs) from the model. The validation of the model is also contained in the model documentation, demonstrating that the model operates according to its design.

Figure 2.2-13 schematically shows a more detailed look at the approach to obtaining information, controlling it in the TSPA database for use in the TSPA-SR model. This database will be controlled within the Technical Data Management System, yet allow access from the TSPA-SR model to obtain input files and run the model. A new procedure, *Verification of Data Entry into the Total System Performance Assessment Database,* LP-IM-001Q-M&O [152182], was developed to catalogue information required by the TSPA-SR model that is housed within Technical Data Management System into a useable form for access by the TSPA-SR model over an electronic connection.

A summary table of software utilized in the TSPA-SR model is presented as Table 2.2-1.

Table 2.2-1. Listing of Software Utilized in the TSPA-SR

Computer Code	Version	STN/CSCI/AMR	Qualification Status	Platform
GoldSim	6.04.007	10344-6.04.007-00	Unqualified	Windows NT 4.0
FEHM	2.1	10086-2.10-00	Unqualified	Windows NT 4.0
T2_BINNING	1.0	MDL-WIS-PA-000002[1]	Qualified	Windows NT 4.0
WT_BINNING	1.0	MDL-WIS-PA-000002[1]	Qualified	Windows NT 4.0
MAKEPTRK	2.0	MDL-WIS-PA-000002[1]	Qualified	Sun OS 5.7
WAPDEG	4.0	1000-4.0-00	Unqualified	Windows NT 4.0
ASHPLUME	1.4LVdll	10022-1.4LVdll-00	Qualified	Windows NT 4.0
SZ_CONVOLUTE	2.0	10207-2.0-00	Qualified	Windows NT 4.0
SEEPDLL	1.0	MDL-WIS-PA-000002[1]	Qualified	Windows NT 4.0
SOILEXP	1.0	MDL-WIS-PA-000002[1]	Qualified	Windows NT 4.0
GVP	1.02	10341-1.02-00	Qualified	Windows NT 4.0
MFD	1.01	10342-1.01-00	Qualified	Windows NT 4.0
SCCD	2.00	10343-2.0-00	Qualified	Windows NT 4.0
PREWAP	1.0	MDL-WIS-PA-00002[1]	Qualified	Windows NT 4.0
MVIEW	2.10	10072-2.10-00	Qualified	Irix 6.3 or greater, HP-UX 10.2, Solaris 2.6, Digital Unix V4
SATOOL	1.0	10084-1.0-00	Qualified	Windows 98
PDFSENS	1.0	10190-1.0-00	Qualified	Windows 98 Windows 95
EQ3/6	7.2b	LLNL:UCRL-MA-110662	Qualified	HP-UXB, 10.20, Windows 98

NOTE: [1]CRWMS M&O 2000 [148384]

Oversight

NRC Technical Exchanges, Appendix 7 Meetings
NWTRB Panel Meetings, Reports to Congress
State of Nevada; Affected Units of Local Government
Public

Prior TSPAs

DOE TSPA-91, 93, 95, VA
NRC IPA-1, -2, -3
EPRI TSPA Phases 1, 2, and 3

Process Model Abstraction

Unsaturated Zone Flow
Engineered Barrier System Environments
Waste Package & Drip Shield Degradation
Waste Form Degradation
Engineered Barrier System Transport
Unsaturated Zone Transport
Saturated Zone Flow and Transport
Disruptive Events
Biosphere

Process Models

Unsaturated Zone Flow Model
Seepage Model
Near Field Geochemistry Model
In-Drift Environment Model
Multi-Scale Thermal Hydrological Model
Waste Package and Drip Shield Corrosion Model
Unsaturated Zone Transport Model
Saturated Zone Flow and Transport Model
Volcanic Eruption Model

Site and Design Information

Site Description Document
Repository Design
Waste Package Design
Laboratory Data
In-Situ Data
Analog Data

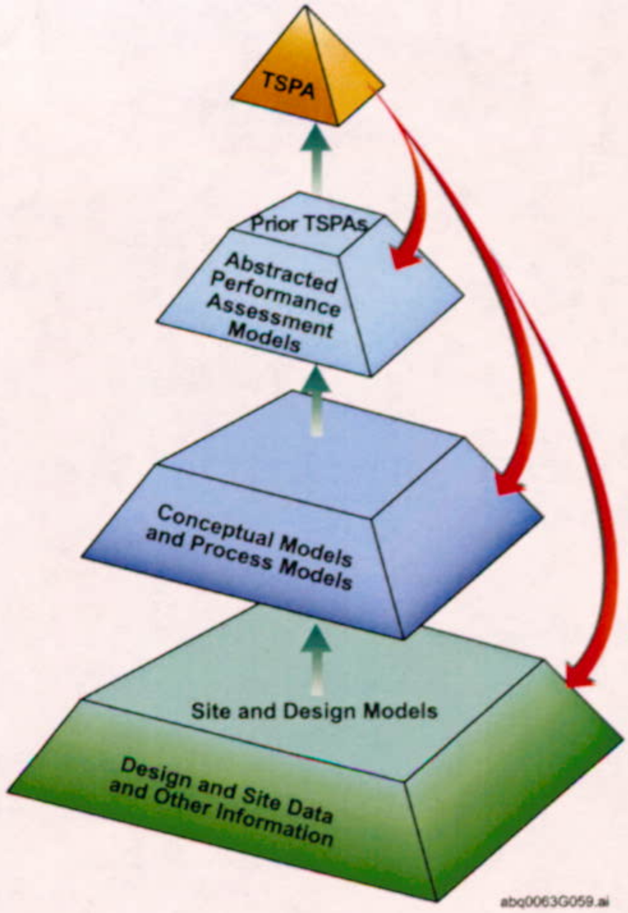

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Figure 2.1-1. Major Sources of Information Used in the Development of the Total System Performance Assessment-Site Recommendation

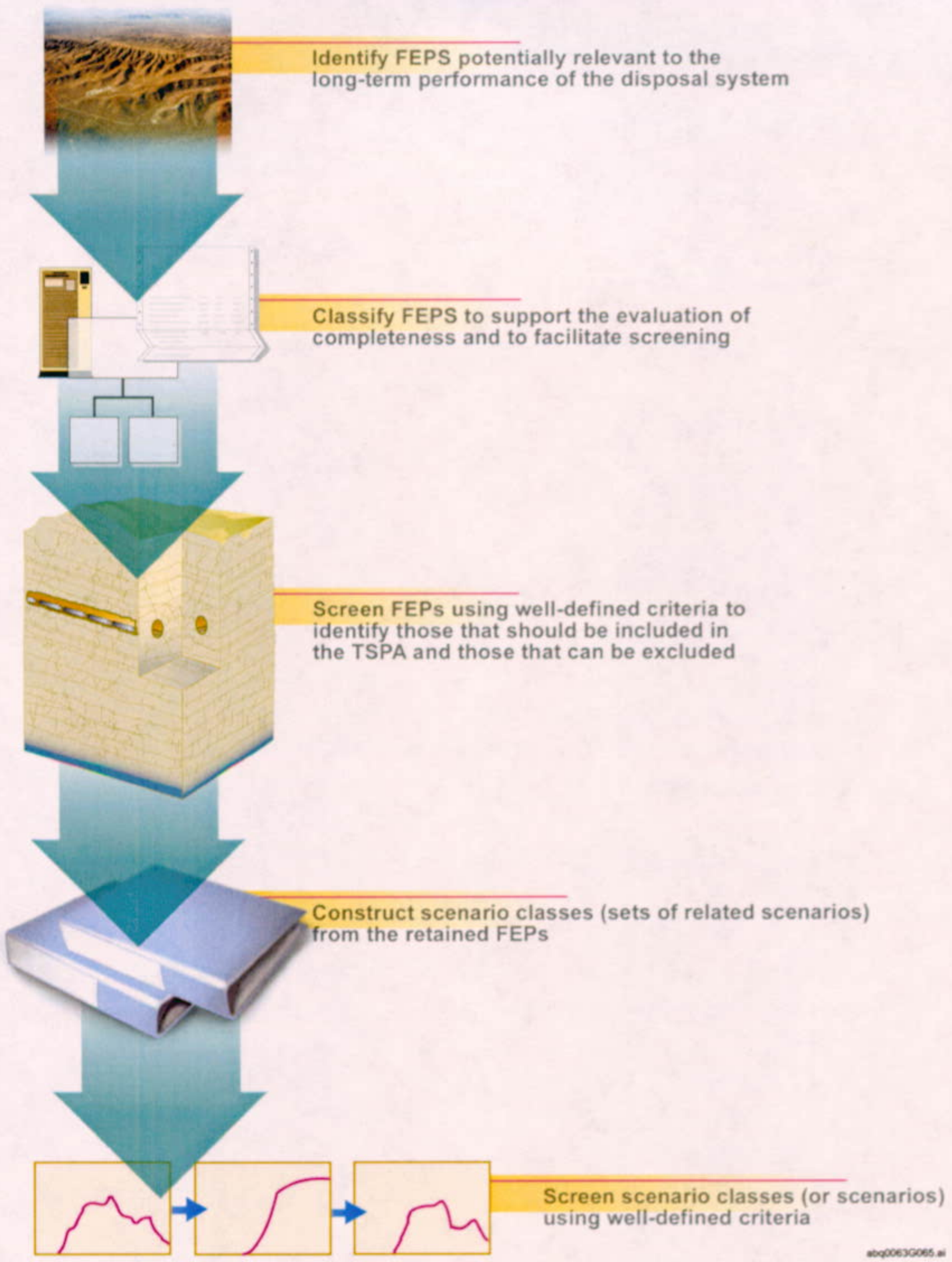

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Figure 2.1-2. The Five Steps in the Formal FEPs Approach for Scenario Development Implemented in Total System Performance Assessment-Site Recommendation

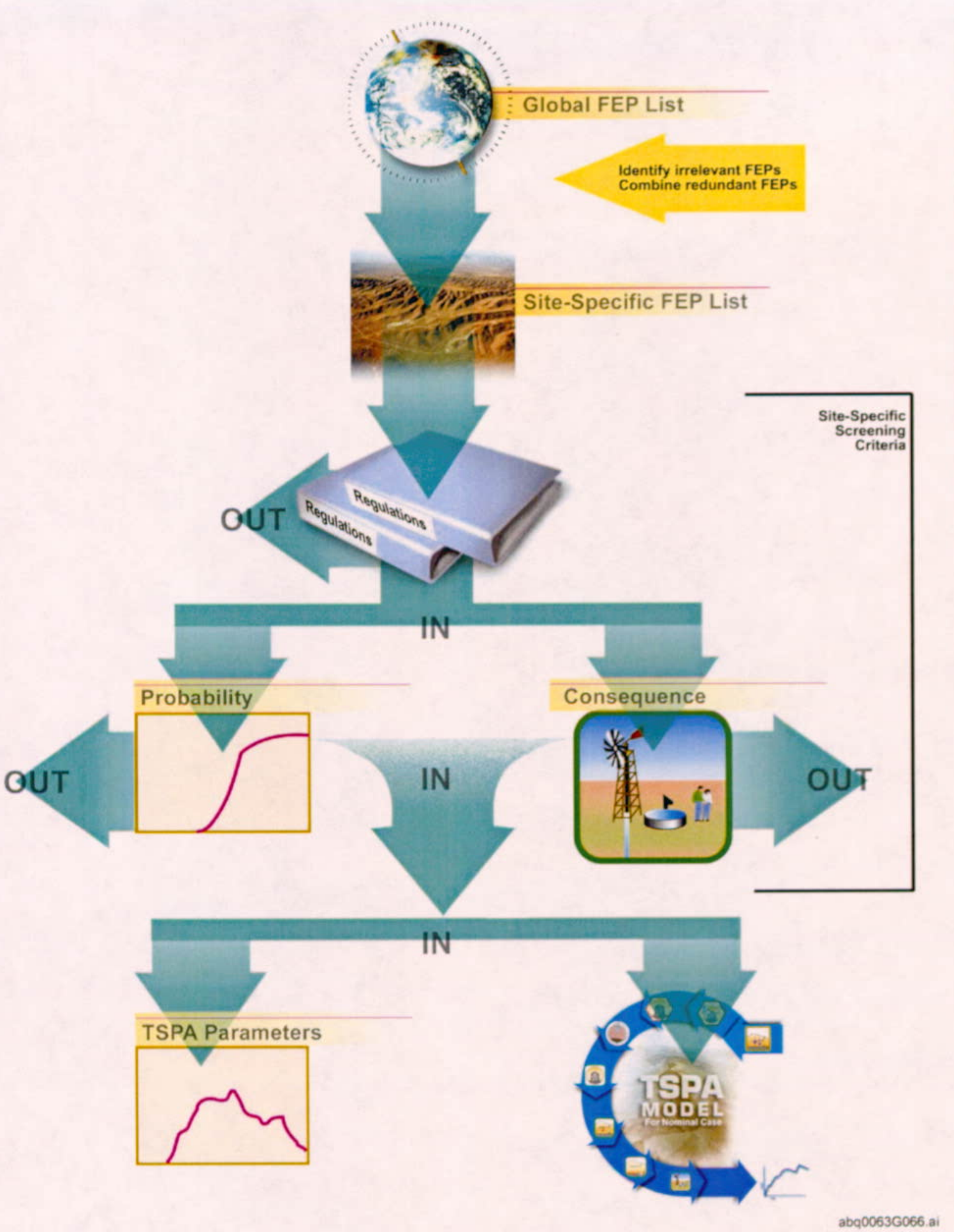

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Figure 2.1-3 Process for Screening FEPs in Total System Performance Assessment-Site Recommendation

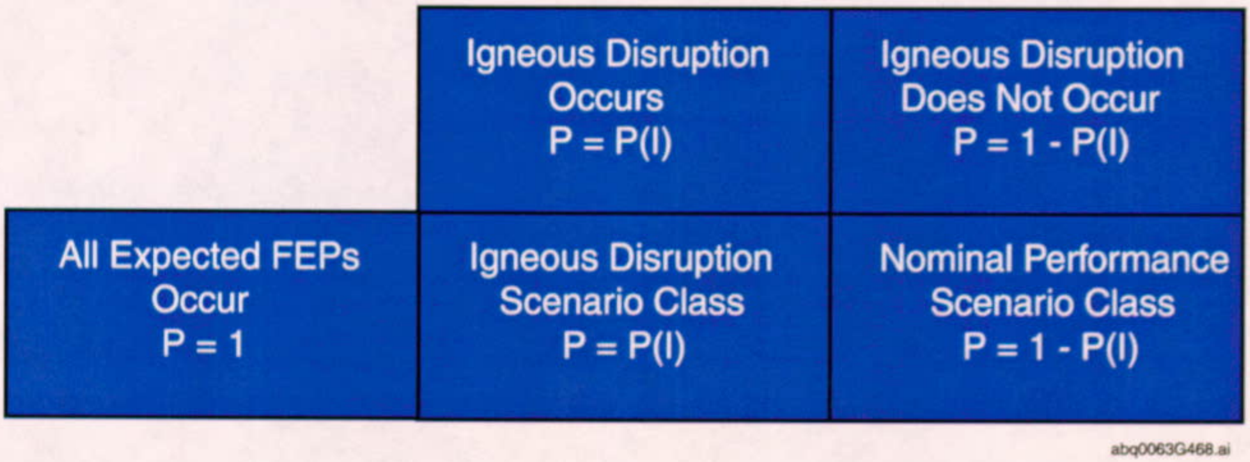

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Figure 2.1-4. Latin Square Scenario Diagram of the Total System Performance Assessment-Site Recommendation Scenario Classes

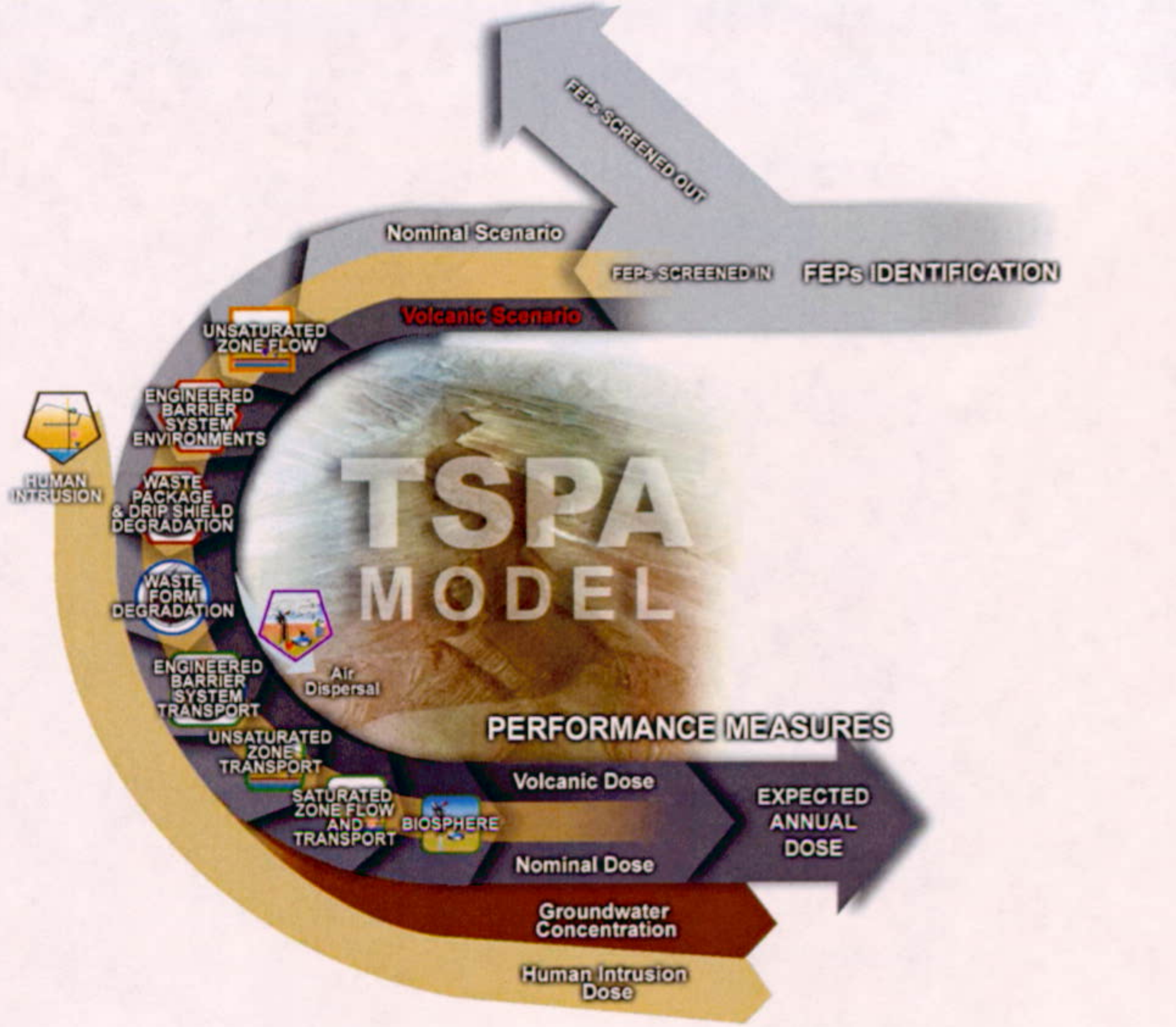

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Figure 2.1-5. Schematic Representation of the Development of Total System Performance Assessment-Site Recommendation Including the Nominal, Disruptive, and Human Intrusion Scenario Classes

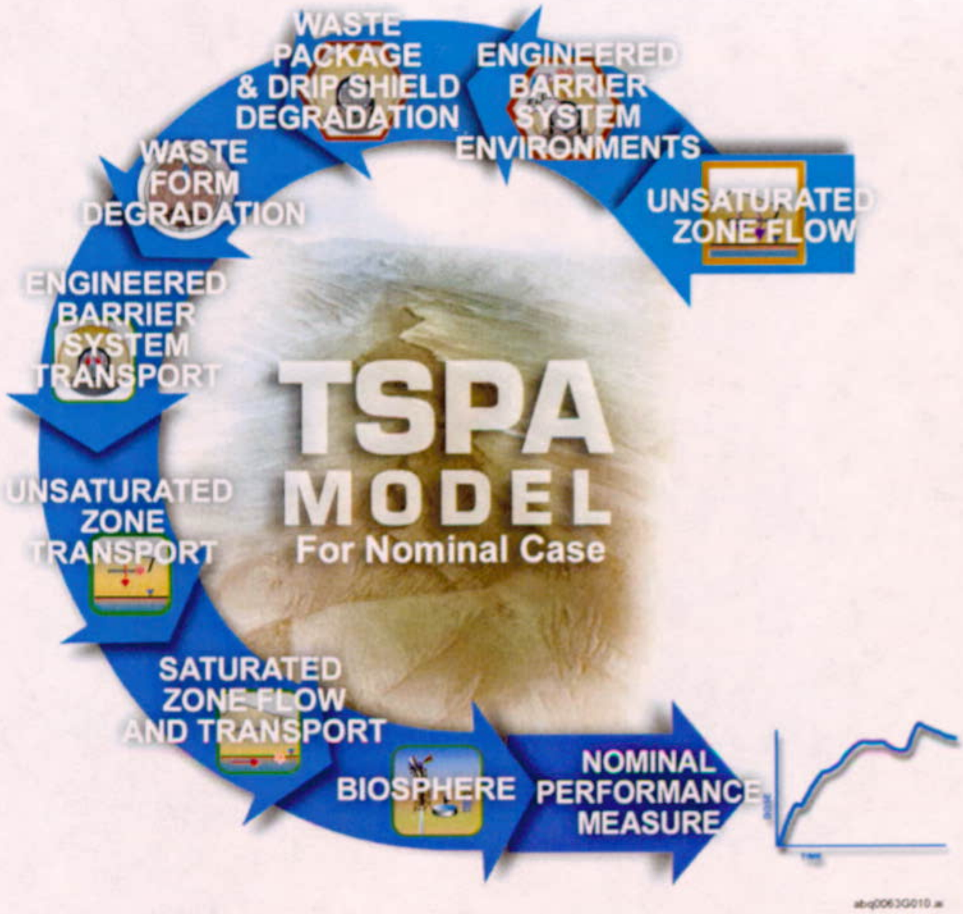

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Figure 2.1-6. Schematic Representation of the Components of the Total System Performance Assessment-Site Recommendation Nominal Scenario Class

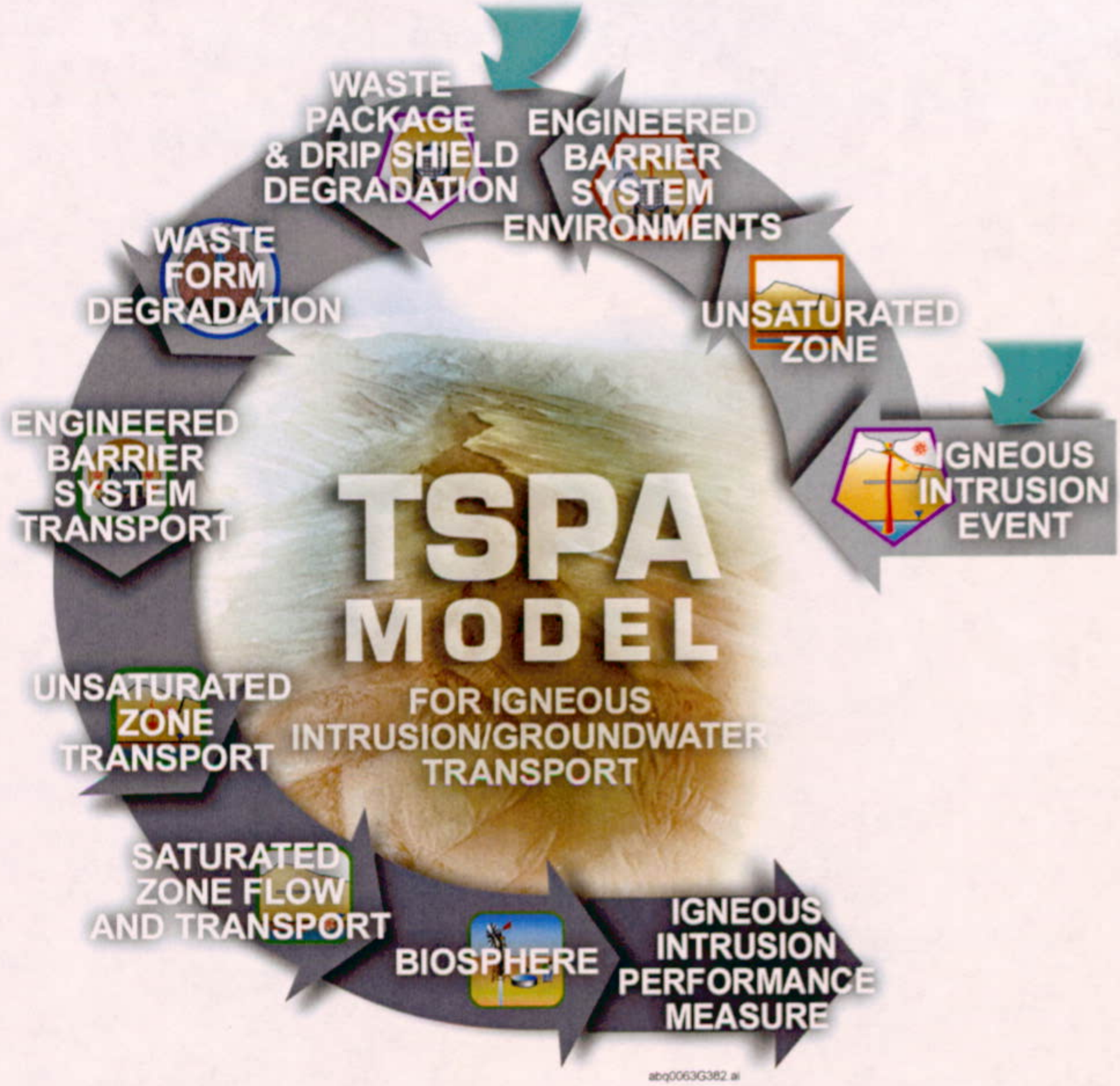

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Figure 2.1-7a. Schematic Representation of the Components of the Total System Performance Assessment-Site Recommendation Disruptive Scenario Class (Igneous Intrusion Scenario)

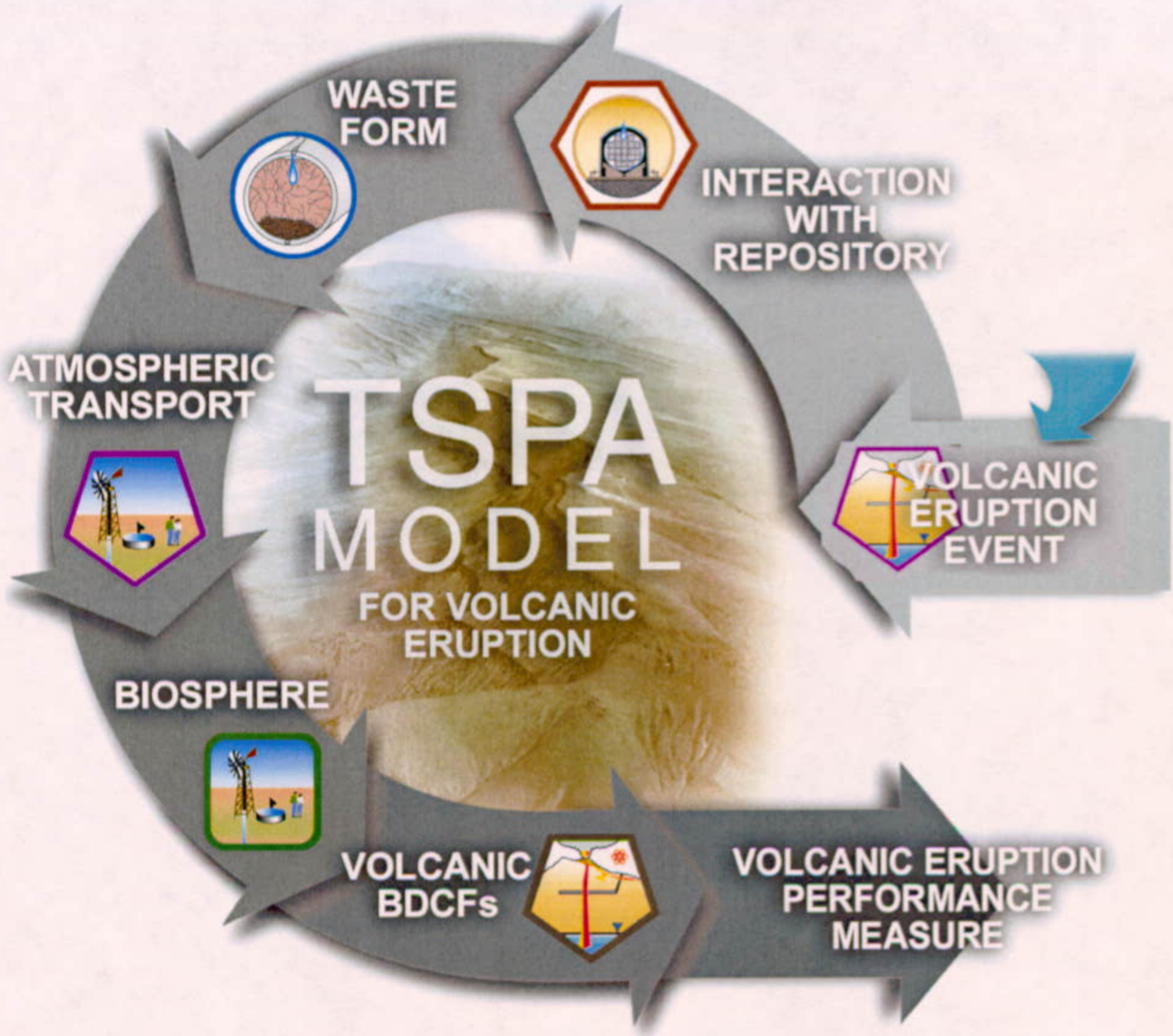

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Figure 2.1-7b. Schematic Representation of the Components of the Total System Performance Assessment-Site Recommendation Disruptive Scenario Class (Volcanic Eruption Scenario)

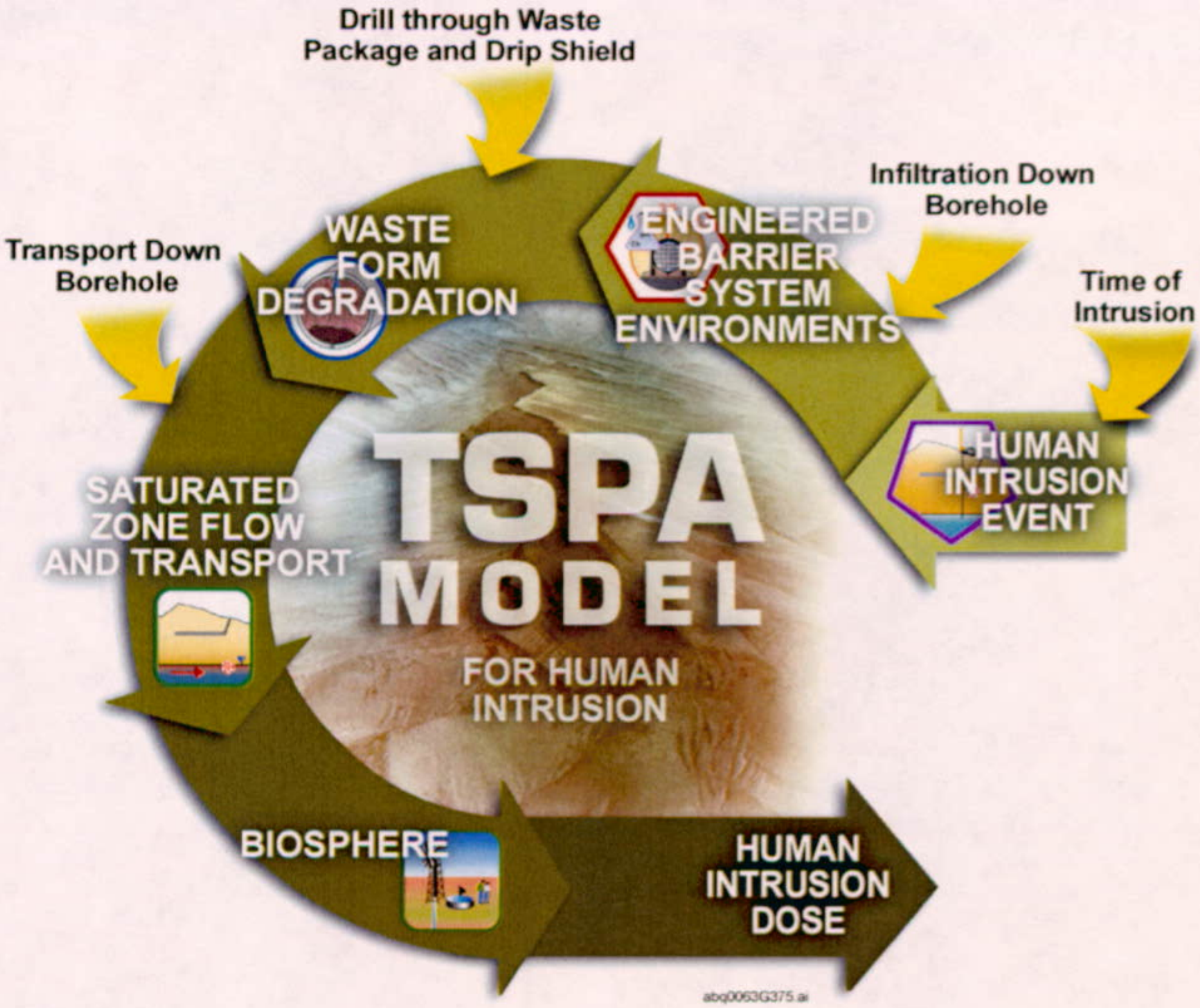

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Figure 2.1-8. Schematic Representation of the Components of the Total System Performance Assessment-Site Recommendation Human Intrusion Scenario Class

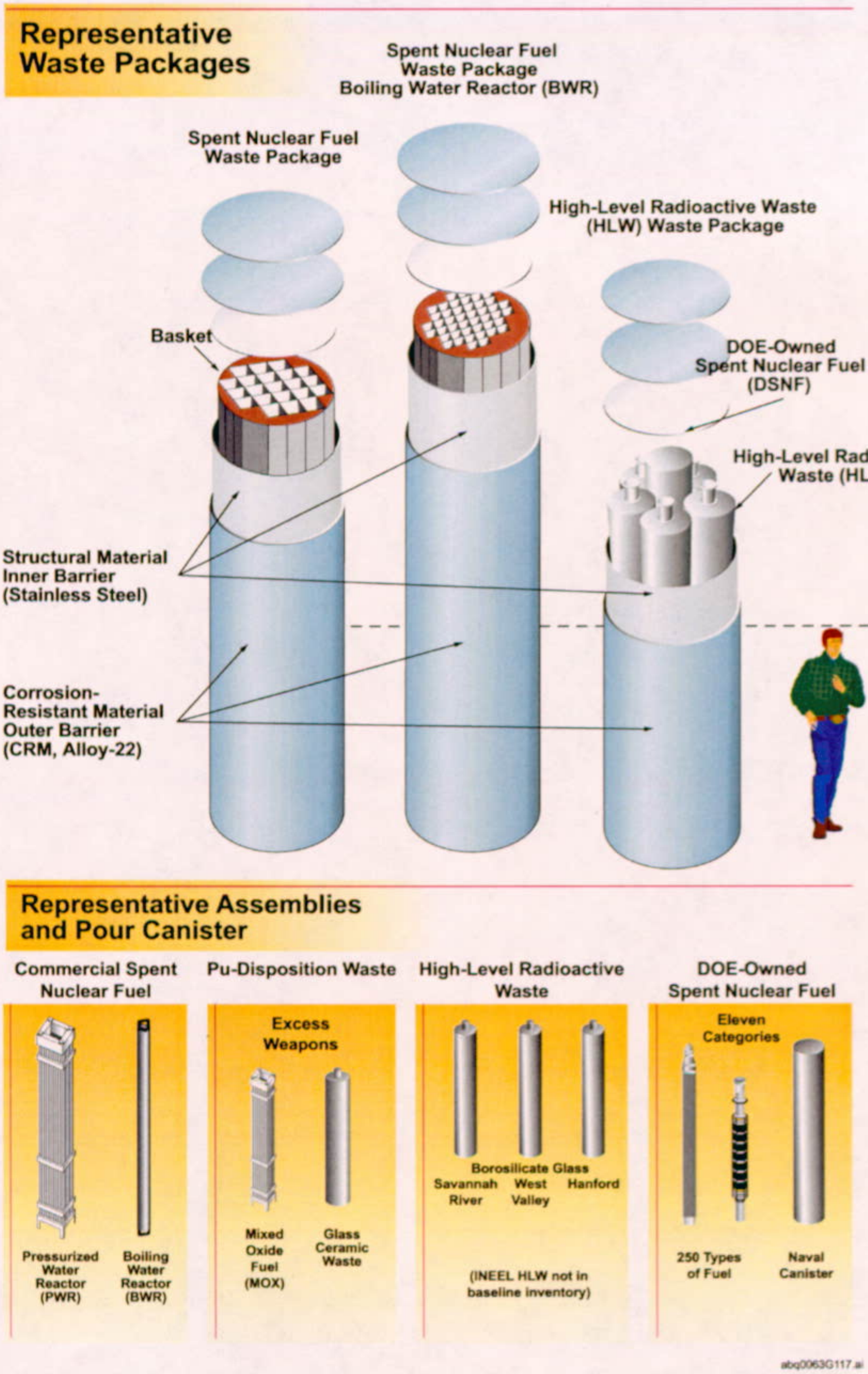

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Figure 2.1-9. Schematic of the Reference Waste Package and Waste Form Designs Used in the Total System Performance Assessment-Site Recommendation

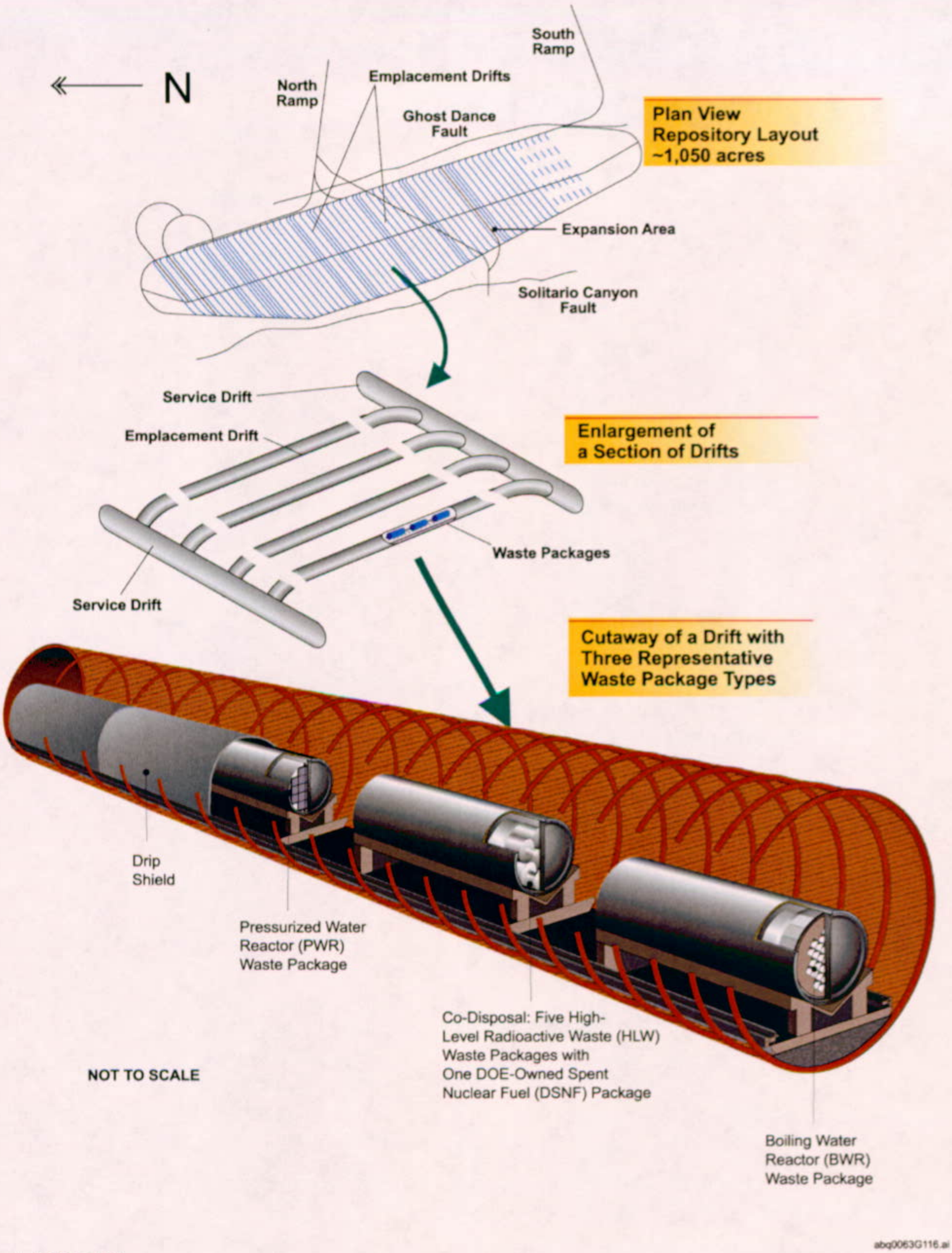

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Figure 2.1-10. Generalized Schematic of Potential Repository System from Mountain Scale to Repository Scale to Waste Package Scale to Waste Form Scale

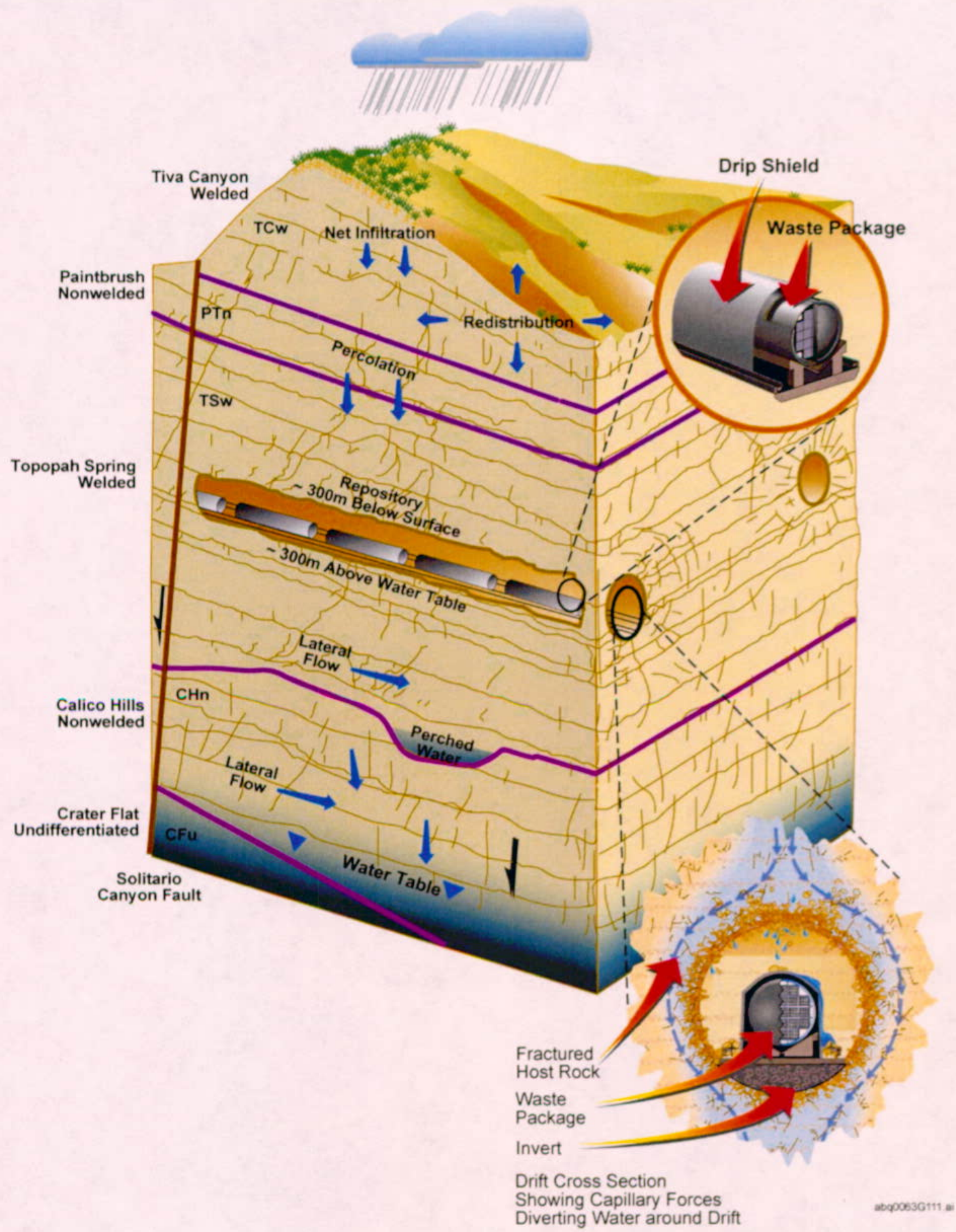

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Figure 2.1-11. Water Movement at Yucca Mountain

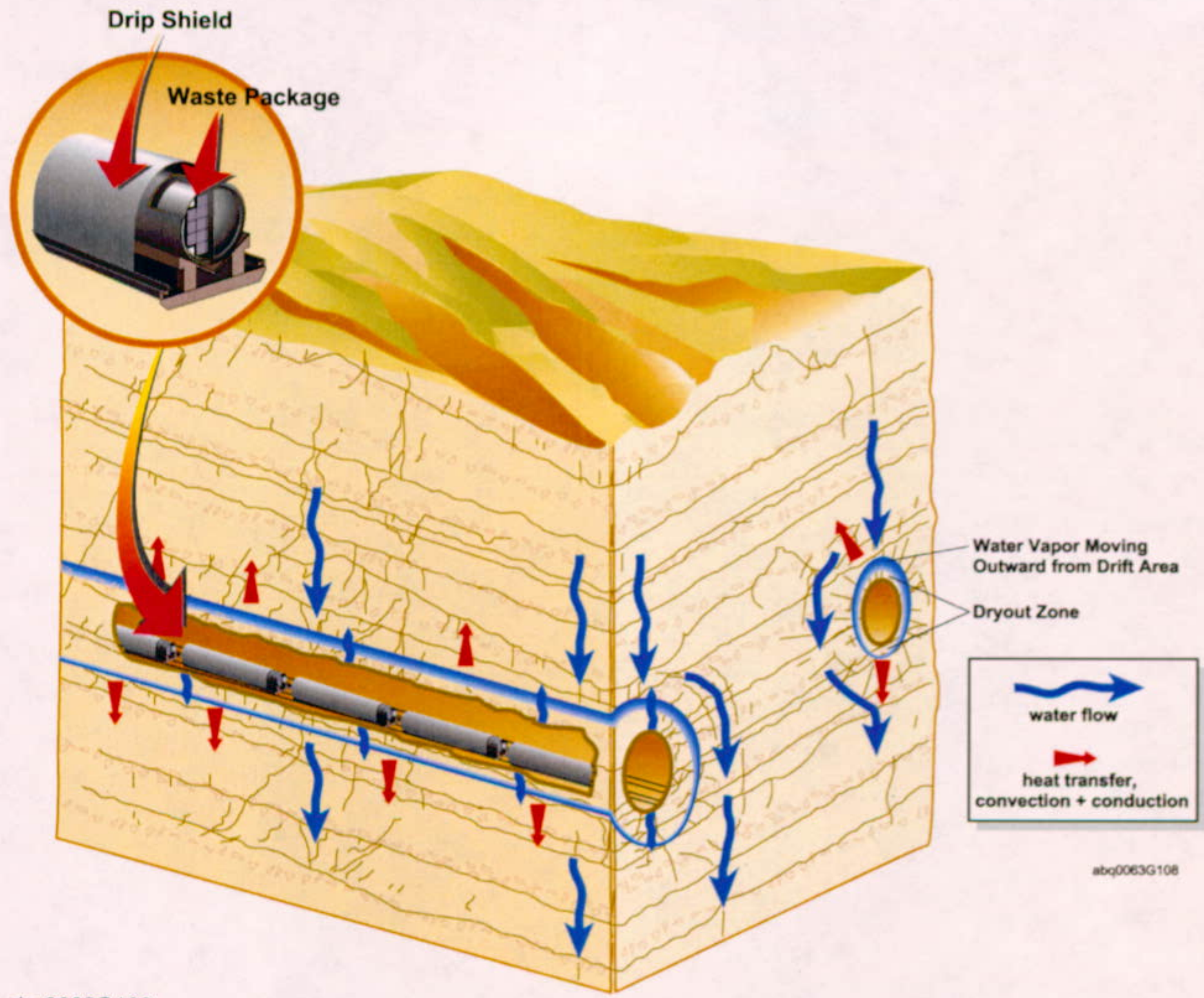

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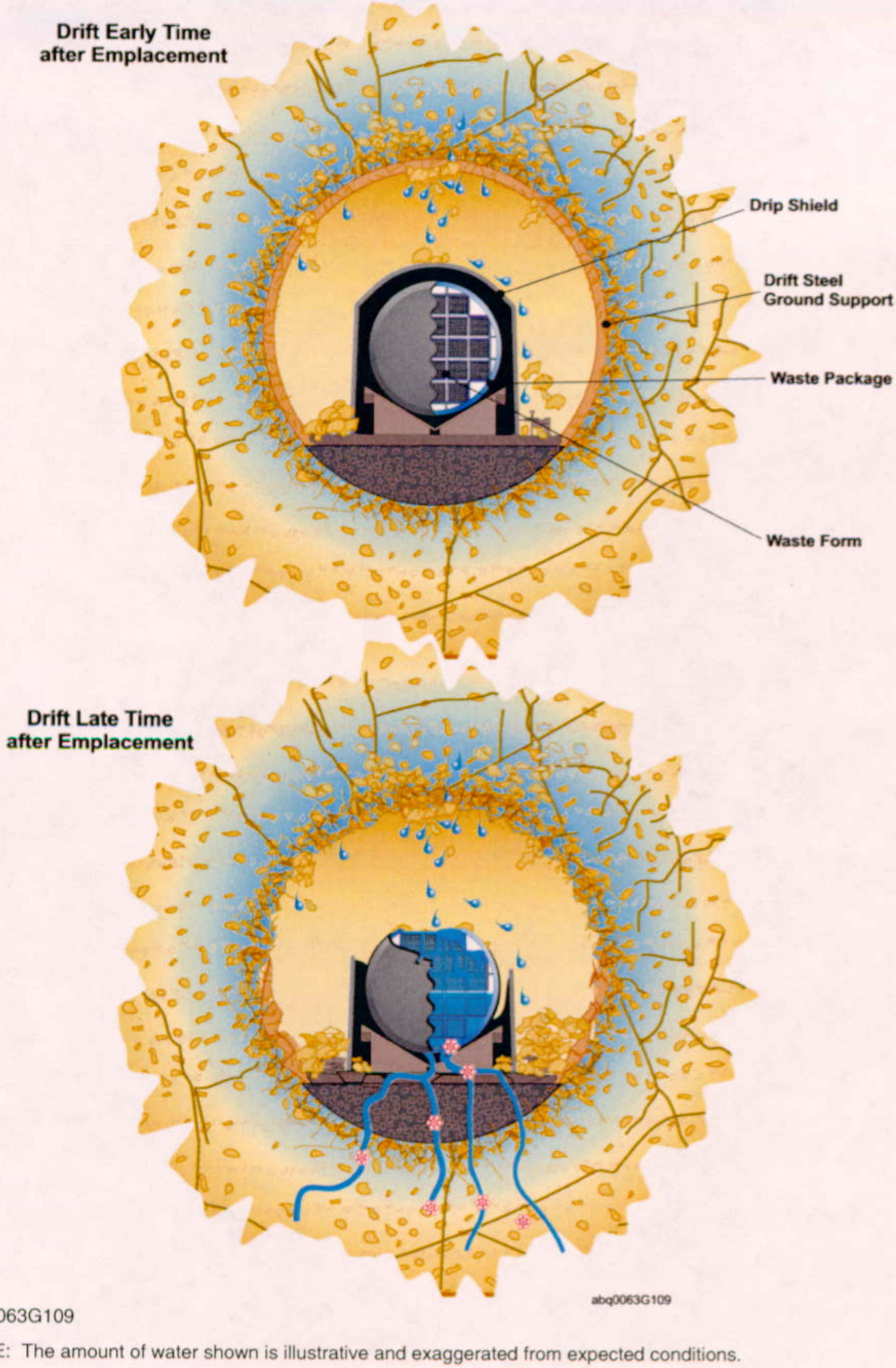
Figure 2.1-12. Water and Vapor Movement at Yucca Mountain Around Drifts



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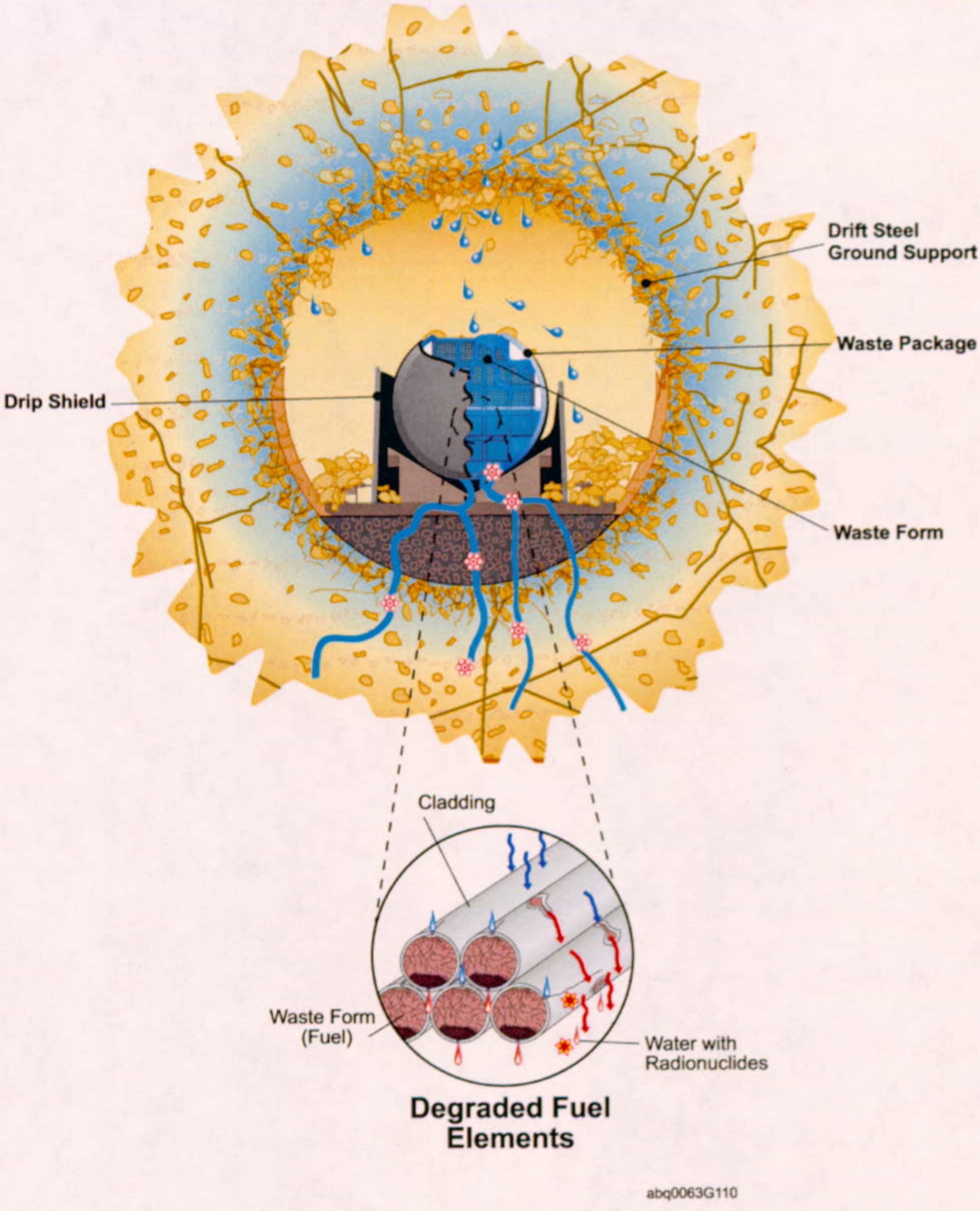
NOTE: The amount of water shown is illustrative and exaggerated from expected conditions.

Figure 2.1-13. Water Movement within Engineered Barrier System



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NOTE: The amount of water shown is illustrative and exaggerated from expected conditions.

Figure 2.1-14. Water Movement and Radionuclide Migration out of Engineered Barrier System

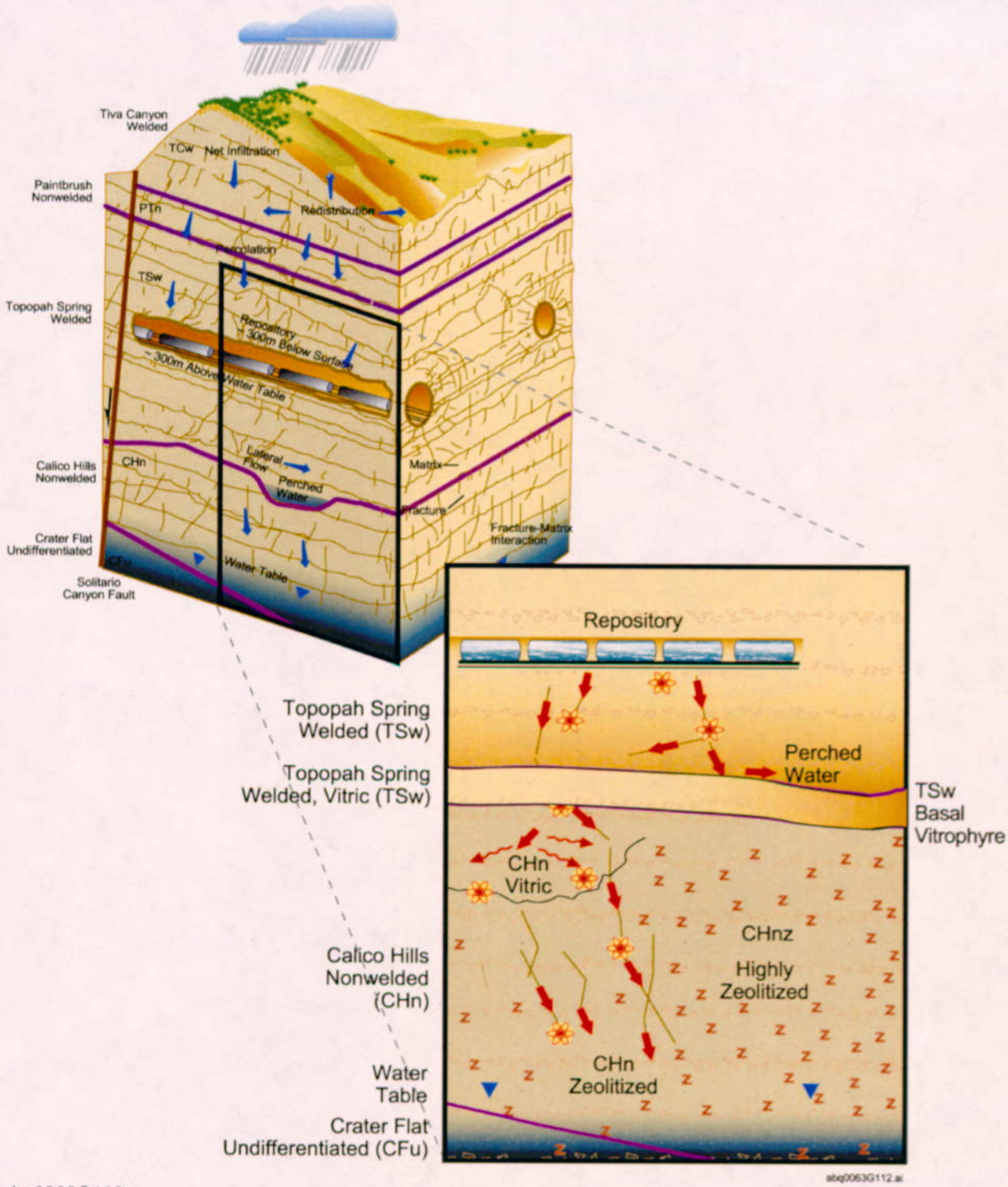

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Figure 2.1-15. Water Movement and Radionuclide Migration through Tuffs

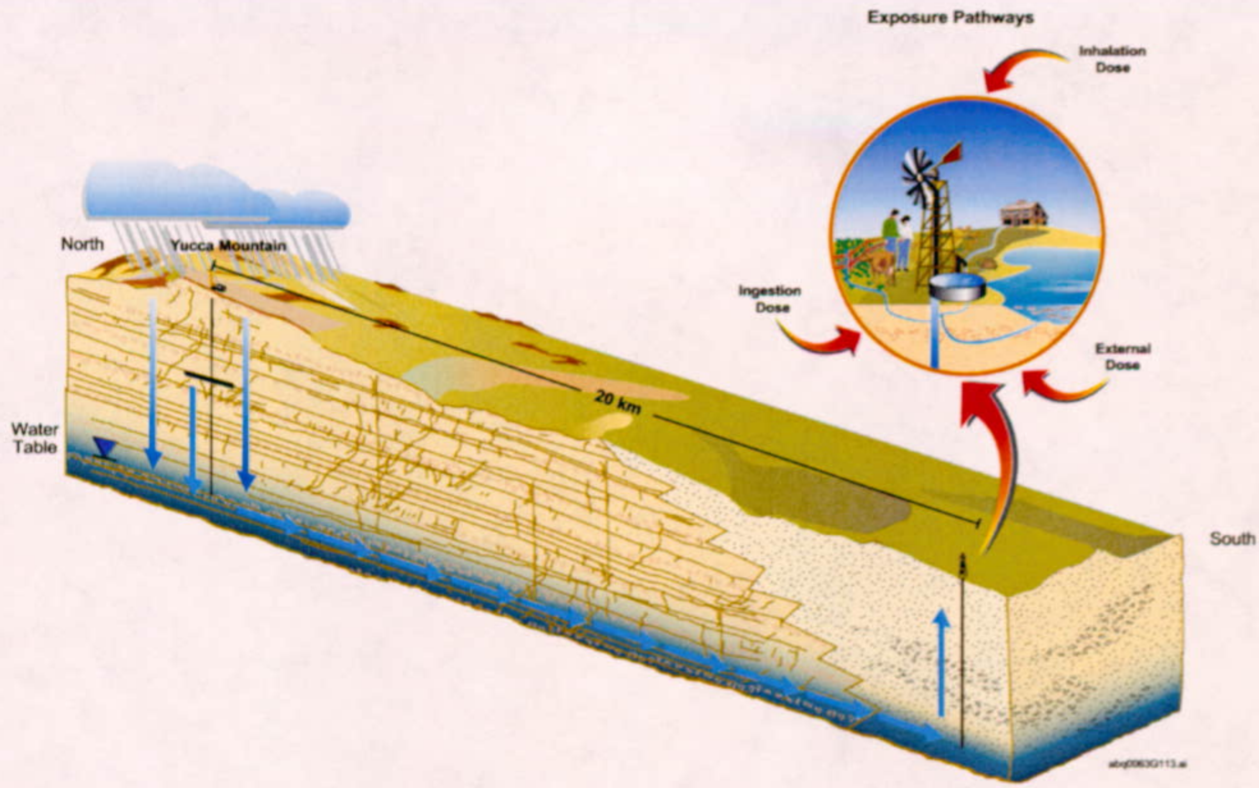

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Figure 2.1-16. Water Movement and Radionuclide Migration through Saturated Zone and Biosphere

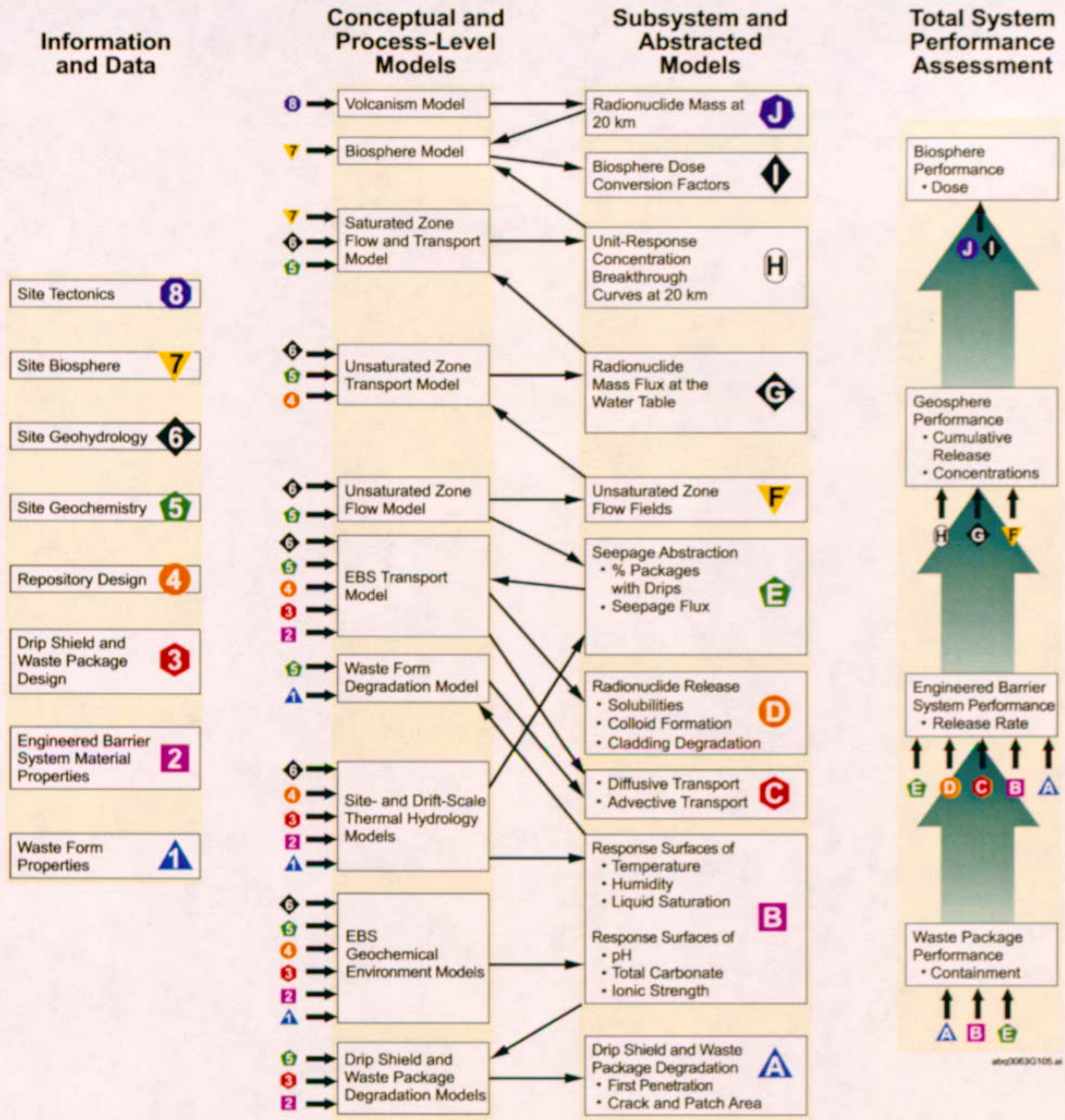

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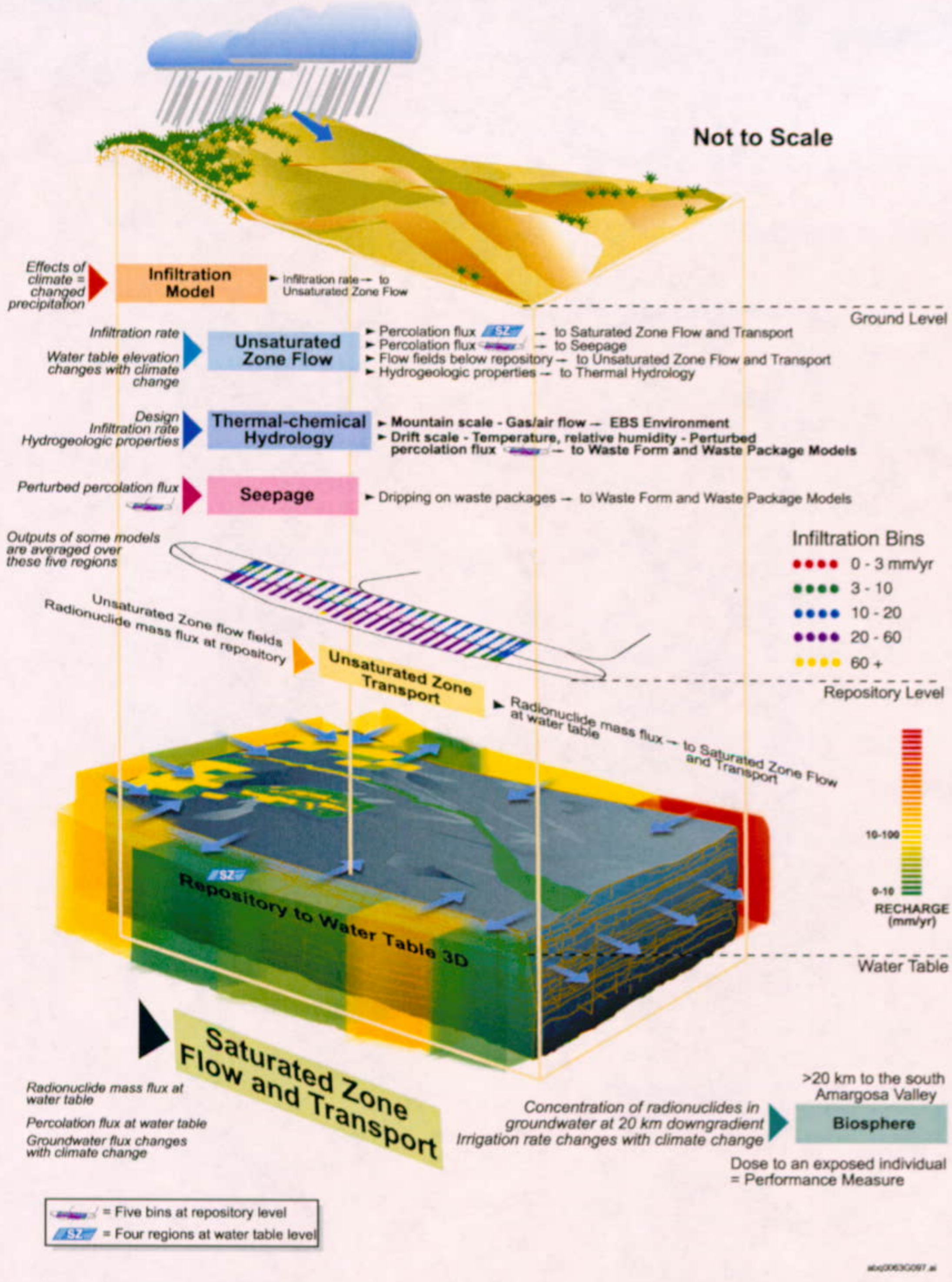
Figure 2.2-1. Simplified Representation of Information Flow in the Total System Performance Assessment-Site Recommendation between Data, Process Models, and Abstracted Models



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NOTE: The Figure is in two parts with the detail of the waste package and waste form models shown in Figure 2.2-2b.

Figure 2.2-2a. Detailed Representation of Information Flow in the Total System Performance Assessment-Site Recommendation

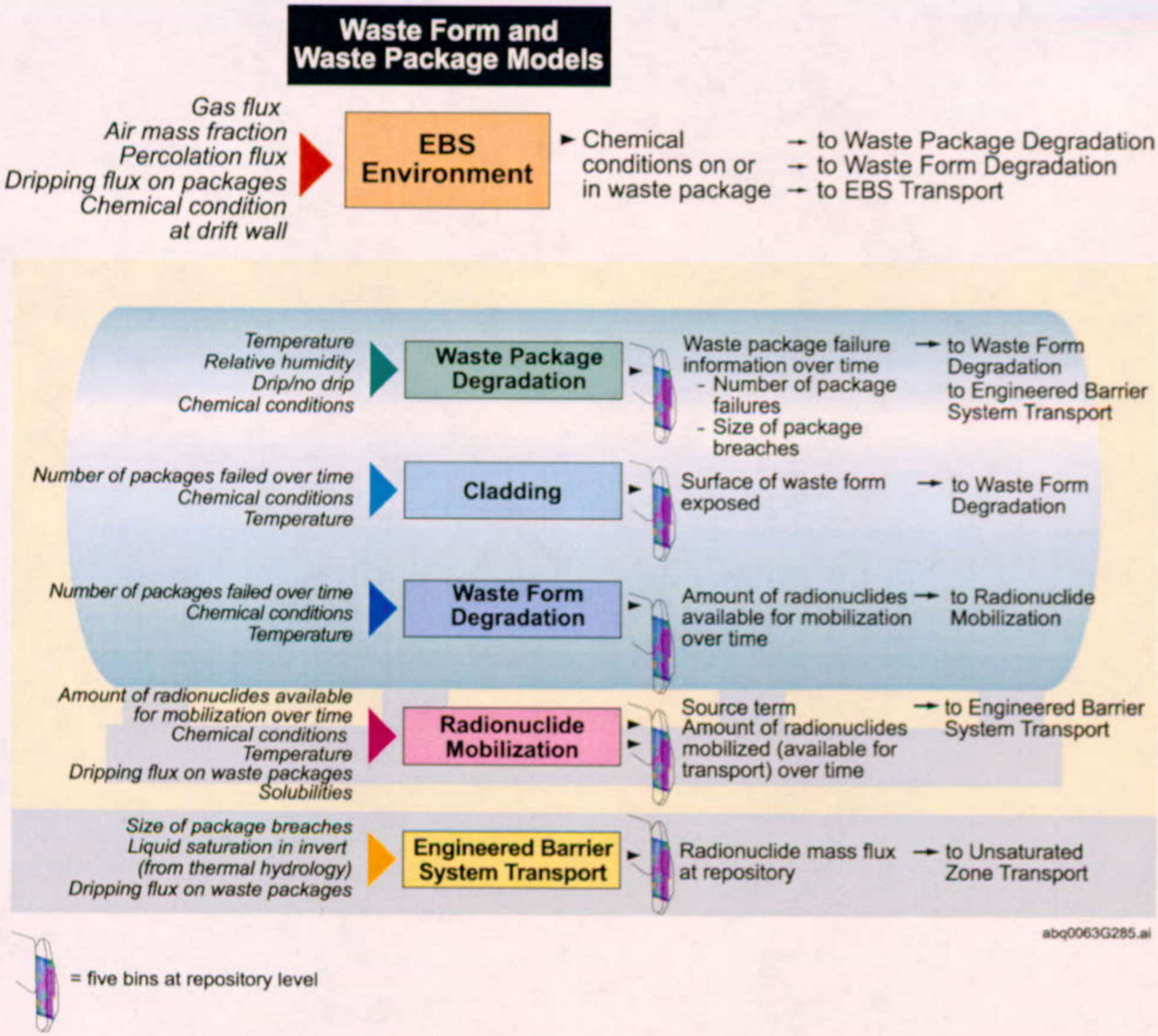

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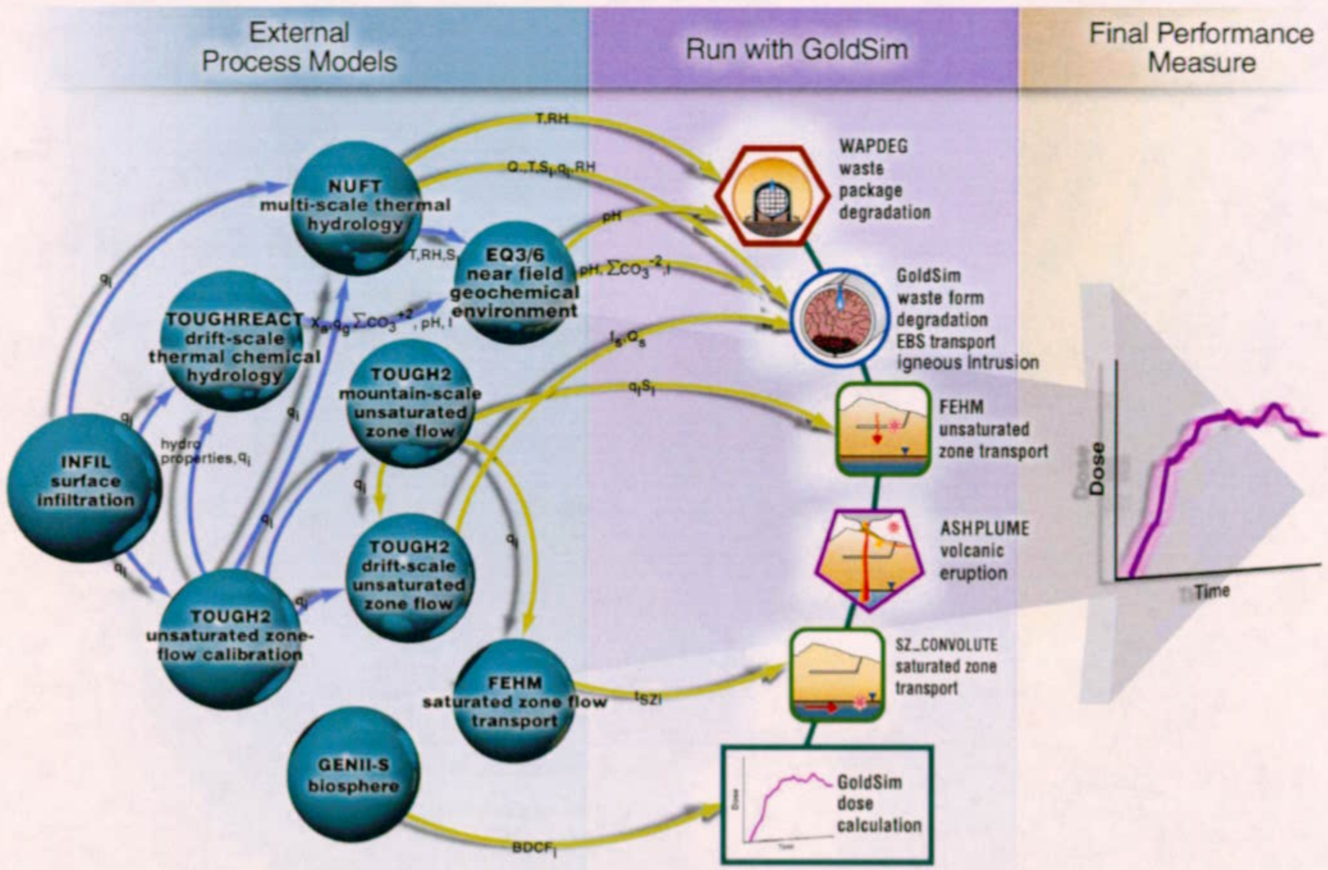
Figure 2.2-2b. Detailed Representation of Information Flow in the Total System Performance Assessment-Site Recommendation



Output Parameters

f_S	Fraction of WPs with Seeps	T	Temperature	I	Ionic Strength
EBS	Engineered Barrier System	RH	Relative Humidity	t_{SZi}	Saturated Zone Transport Time
Q_S	Seep Flow Rate	S_l	Liquid Saturation	$BDCF_i$	Biosphere Dose Conversion Factor
$Q_{\cdot}$	Evaporation Rate	X_a	Air Mass Fraction	q_g	Gas Flux
pH	pH	q_l	Liquid Flux		
ΣCO_3^{-2}	Carbonate Concentration	q_i	Infiltration Flux		

Legend

Response Surface Between Process Models

Response Surface from Process Model to GoldSim

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Figure 2.2-3. Total System Performance Assessment-Site Recommendation Code Configuration: Information Flow Among Component Computer Codes

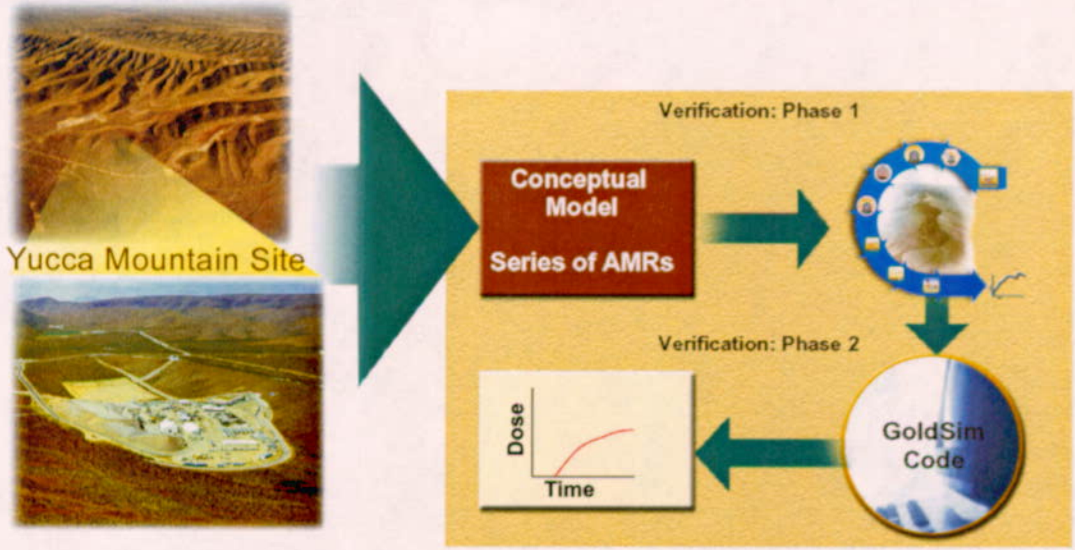

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Figure 2.2-4. Testing of Integrated Total System Performance Assessment-Site Recommendation Model

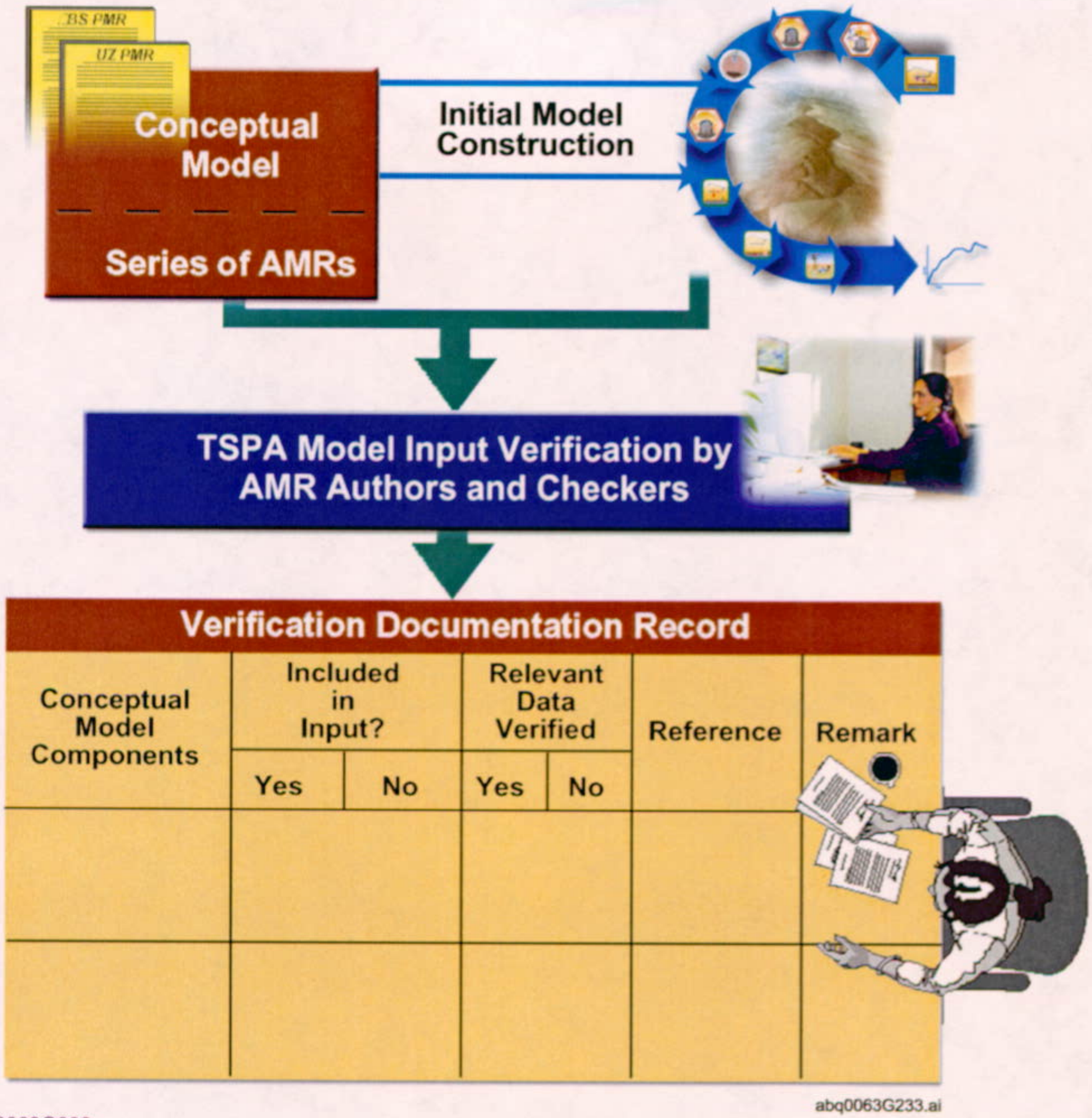

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Figure 2.2-5. Phase 1: Verification of Total System Performance Assessment-Site Recommendation Model

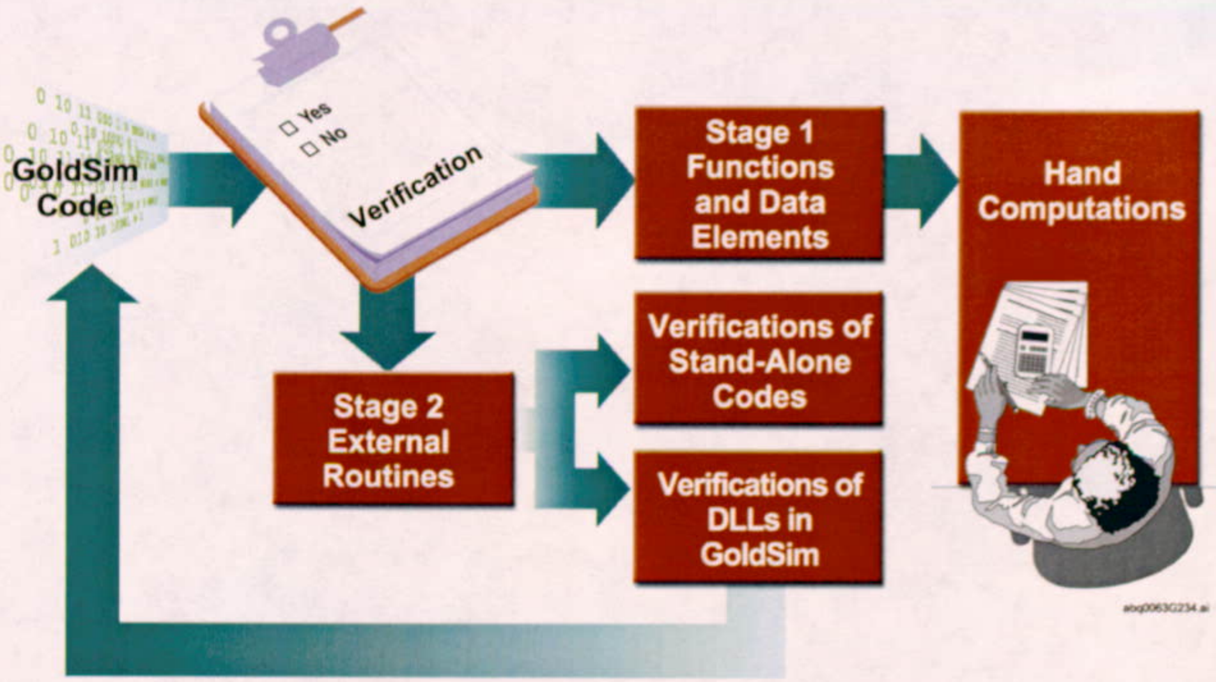

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Figure 2.2-6. Phase 2: Verification of Total System Performance Assessment -Site Recommendation Model (Stages 1 and 2)

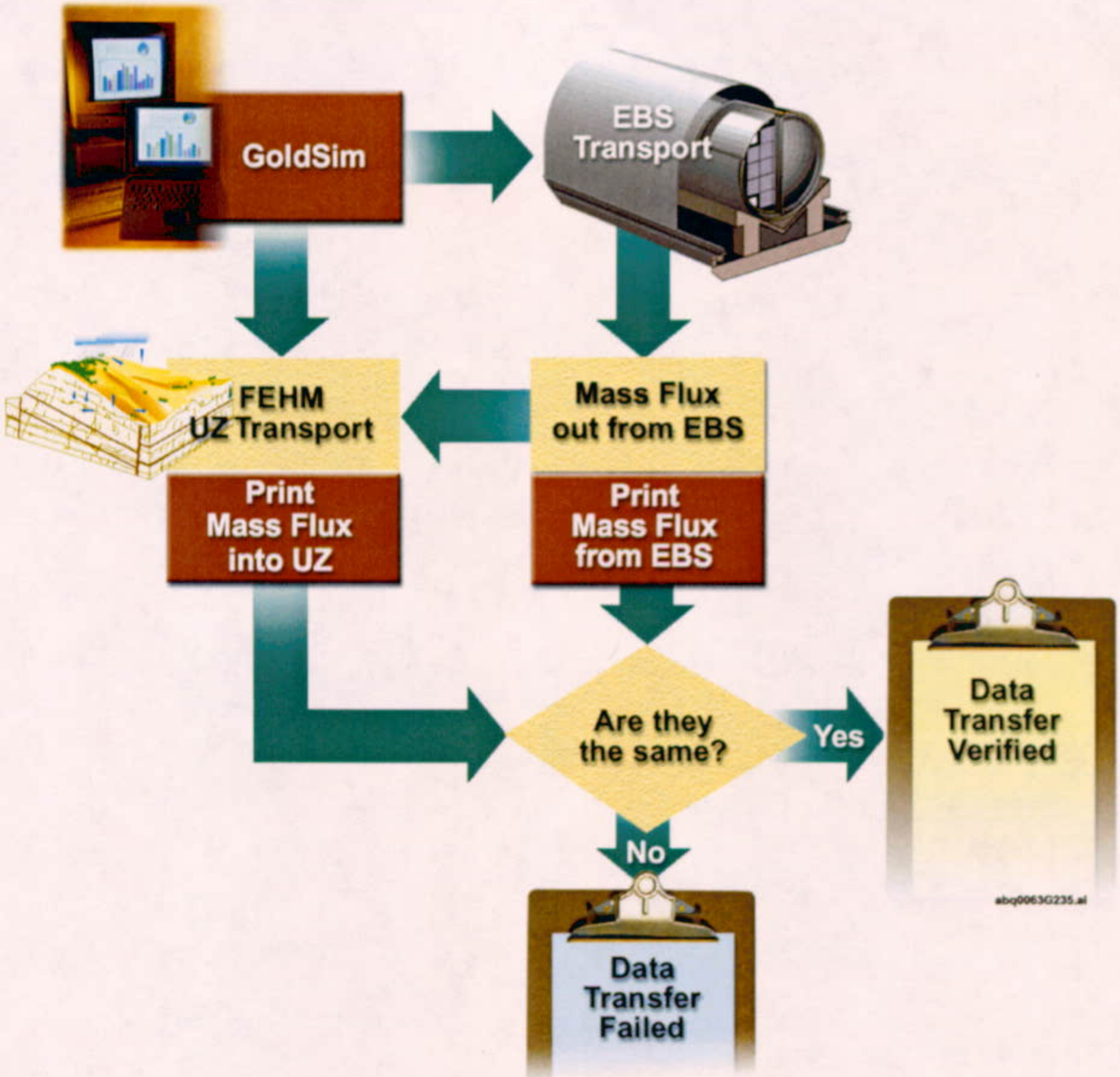

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Figure 2.2-7. Phase 2: Verification of Total System Performance Assessment -Site Recommendation Model (Stage 3)

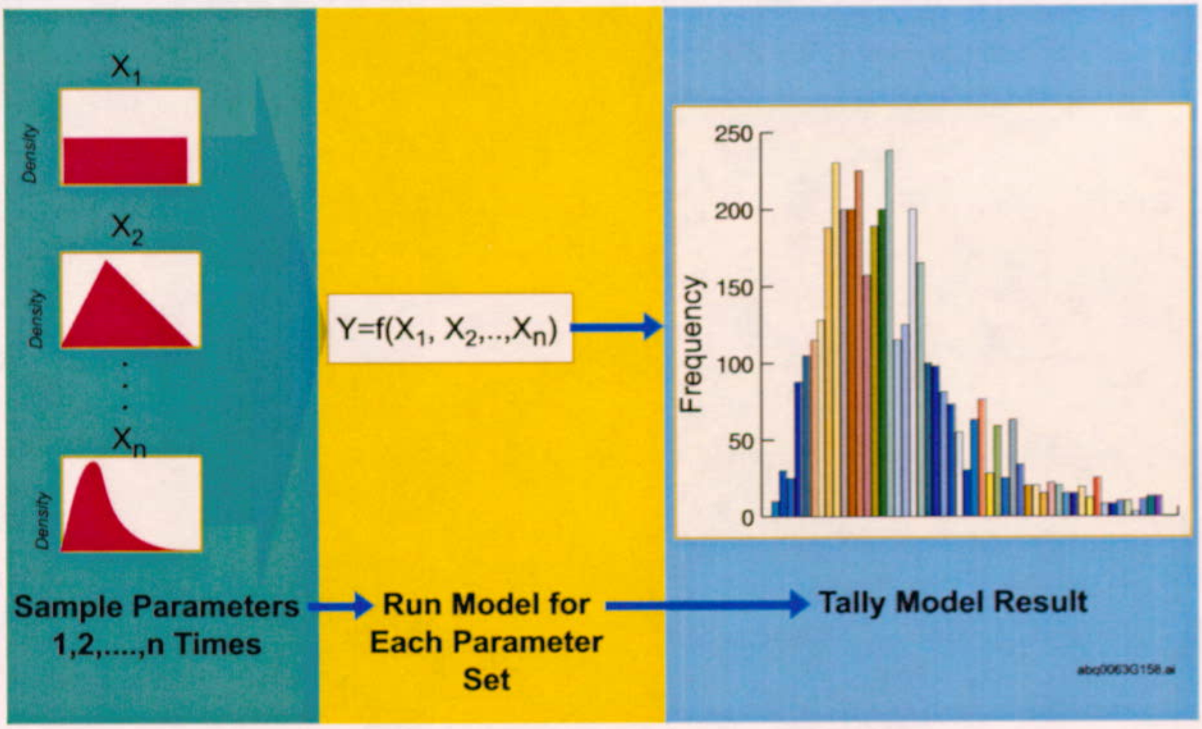

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Figure 2.2-8. Schematic of Monte-Carlo Simulation Methodology

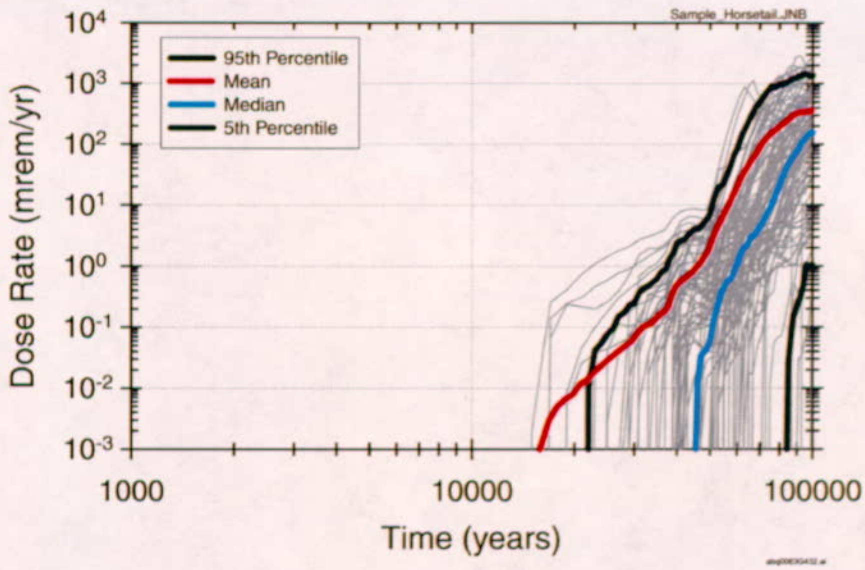

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Figure 2.2-9. Format for Presenting Probabilistic Model Results in Total System Performance Assessment-Site Recommendation

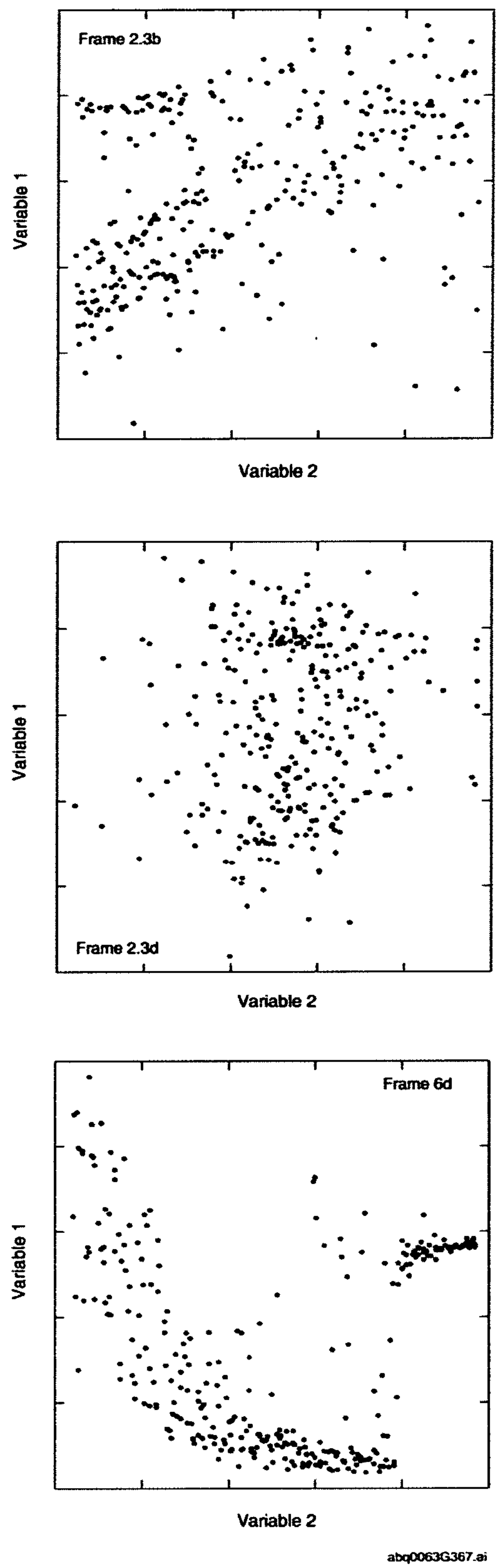

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Figure 2.2-10. Example Scatter Plots

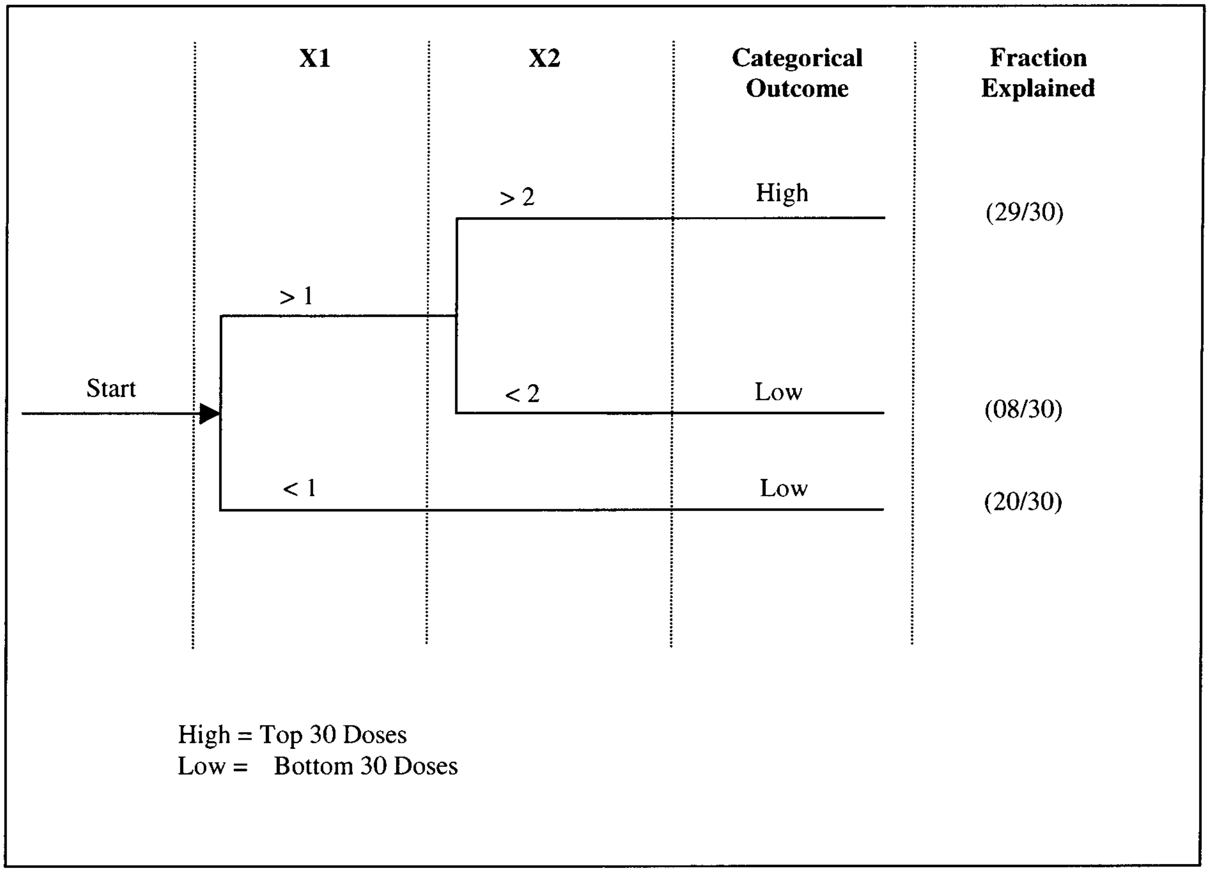

Figure 2.2-11. Example Binary Decision Tree

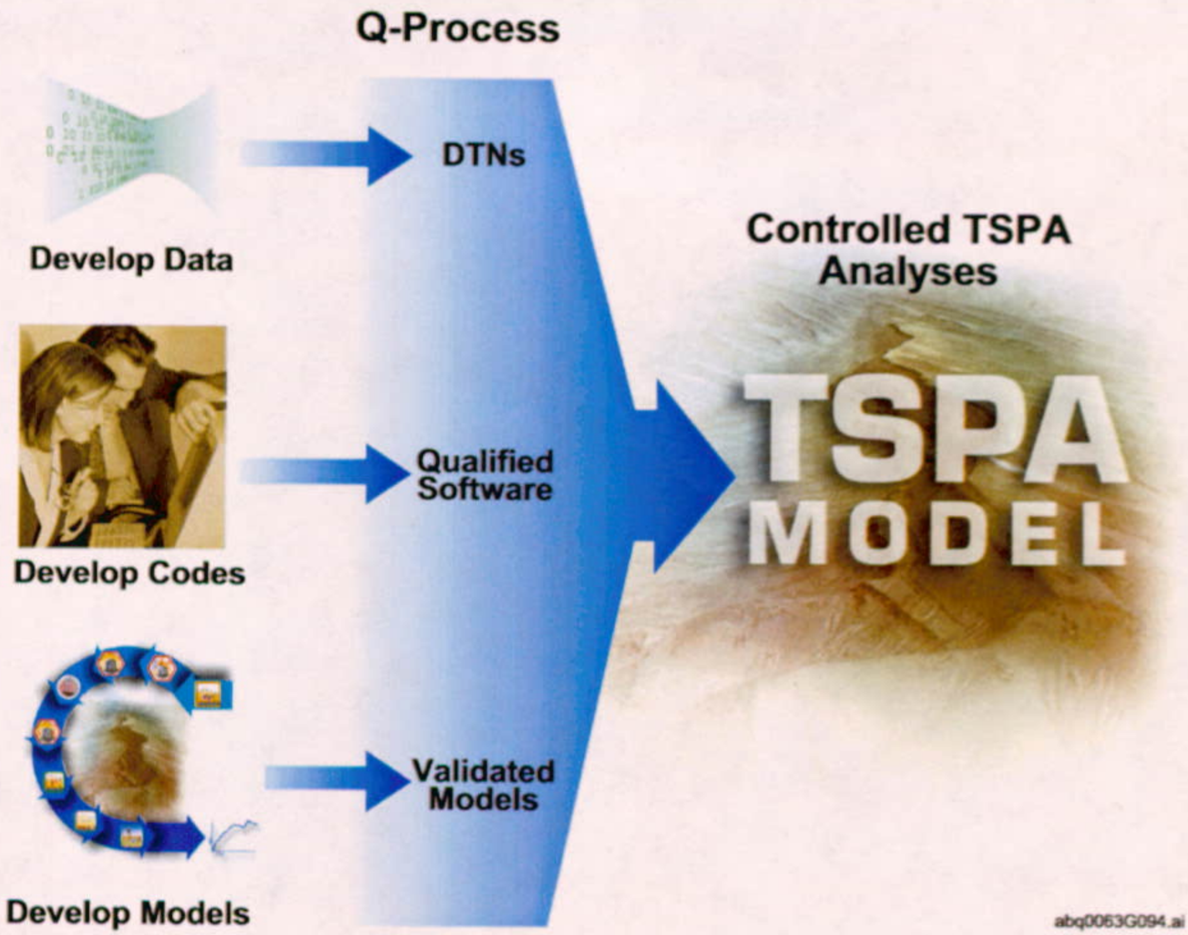

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Figure 2.2-12. Control of Total System Performance Assessment Model Development and Analyses

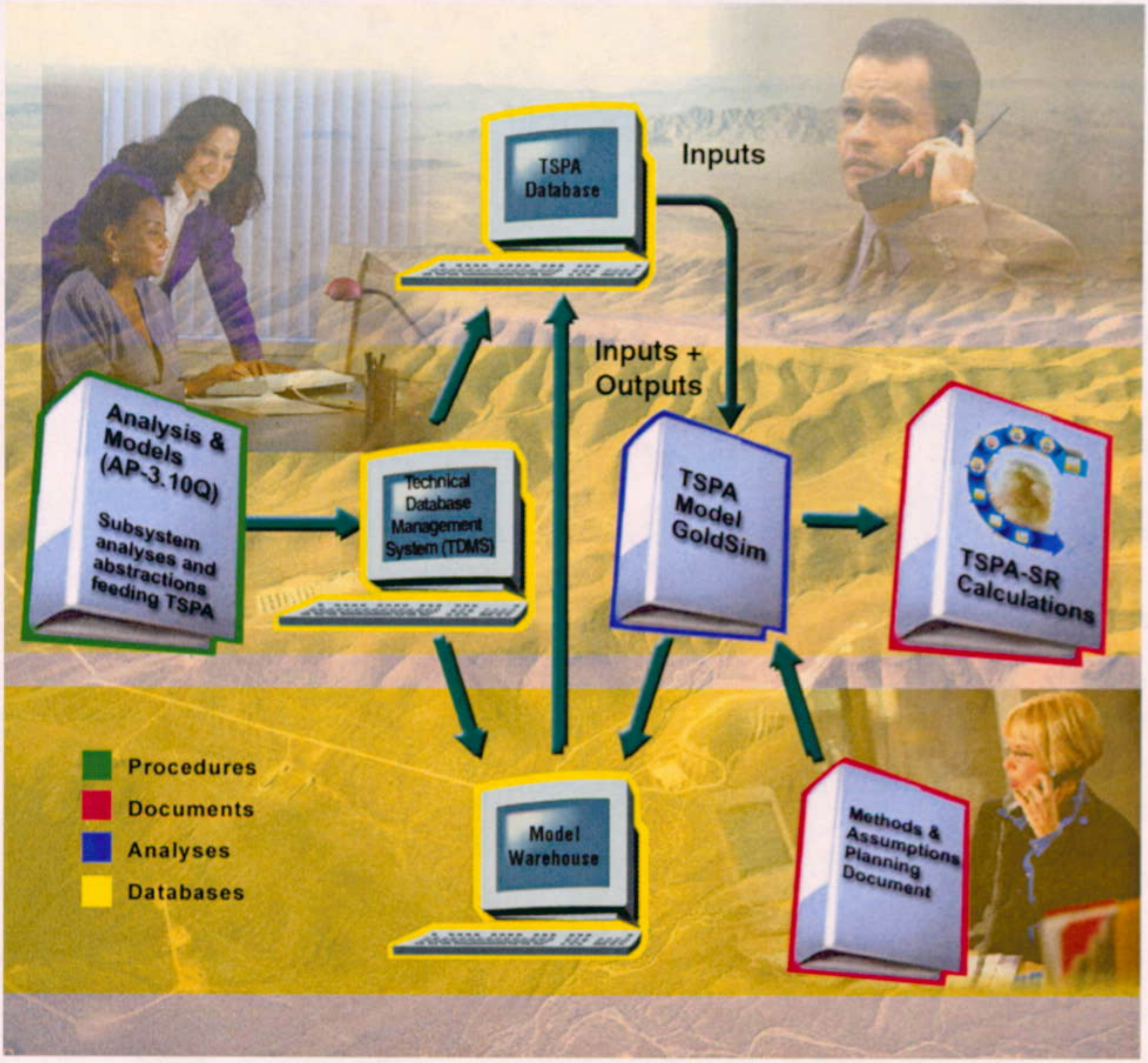

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Figure 2.2-13. Control of Information Flow into Total System Performance Assessment-Site Recommendation Model

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3. DEVELOPMENT OF TOTAL SYSTEM PERFORMANCE ASSESSMENT

The Yucca Mountain system has been divided into individual parts to make the overall system analyses manageable. Each of the individual parts or components represents a major process area. The component models that define the process areas are discussed in this section. The TSPA-SR model document (CRWMS M&O 2000 [148384]) provides detail as to how the model is implemented in the TSPA-SR model, and the PMRs describe the models and their associated Analysis Model Reports (AMRs). In the AMRs, the rationale for carrying forward a particular component model as opposed to an alternative model is documented. If the model is included in the TSPA-SR and described in this section, it has been determined to be the most defensible model of the process. In many cases, the models have been evaluated in the AMR and determined to be most important to subsystem performance of the particular process, though it's significance to overall performance is not known until it is embedded in the TSPA model. Thus, this document does not go into extensive discussion of alternative conceptual models, unless the AMR has identified that the TSPA should explicitly evaluate more than one model for a particular component of the system. Additional evaluation of alternative conceptual models that are consistent with the available data are discussed in the supporting AMRs and summarized in the applicable PMRs indentified in Table 3-1.

The process areas are listed in Table 3-1, along with the corresponding section number in the TSPA-SR, and the associated PMR. The disruptive event, volcanism, is discussed in the last subsection of this section. In addition to the process areas discussed in this section, the effects of human intrusion on potential repository performance is discussed in Section 4.

Table 3-1. Component Models for Process Areas and Corresponding Documentation

Process Area	TSPA-SR Section	PMR Documentation
UZ Flow	3.2	UZ Flow and Transport (CRWMS M&O 2000 [145774])
Engineered Barrier System Environments	3.3	Near Field Environment (CRWMS M&O 2000 [153178])
Waste Package and Drip Shield Degradation	3.4	Waste Package Degradation (CRWMS M&O 2000 [138396])
Waste Form Degradation	3.5	Waste Form Degradation (CRWMS M&O 2000 [138332])
Engineered Barrier System Transport	3.6	Engineered Barrier System Degradation, Flow, and Transport (CRWMS M&O 2000 [145796])
UZ Transport	3.7	UZ Flow and Transport (CRWMS M&O 2000 [145774])
SZ Flow and Transport	3.8	SZ Flow and Transport (CRWMS M&O 2000 [145738])
Biosphere	3.9	Biosphere (CRWMS M&O 2000 [151615])
Volcanism	3.10	Disruptive Events (CRWMS M&O 2000 [141733])

NOTE: Some PMRs also contributed to other process areas.

The various TSPA component models are represented using different computer codes, previously described in Section 2. The variations in the codes, in terms of their architecture, require that the form of the input and output parameters meet the specific code requirements. For this reason, the different models may require somewhat different forms of what would appear to be the same input and output parameters. The specific parameters used in the TSPA-SR are given in the TSPA-SR model document (CRWMS M&O 2000 [148384]).

Although the processes will be interrelated in the actual repository system, the assumption is that the components could be treated separately if a consistent set of boundary conditions and scenarios is rigorously maintained among all the related components.

This section addresses the major features, processes, and conceptualization of the nine process areas as well as the implementation of these components into the performance assessment analyses. It also provides the results and interpretations of the analyses associated with the component. Section 4 describes how these components were recombined into the total system model, and Section 5 discusses the sensitivity of the total-system results to various aspects of these components.

3.1 INTRODUCTION

As noted in Sections 1.6 and 2.1, the process of developing an integrated analysis of potential repository performance requires the construction of an integrated model that includes all relevant features, events and processes and their quantitative description within individual component models. This integrated set of models is presented in the various sections of Section 3.

The purpose of this introductory section of Section 3 is to provide the reader with a series of roadmaps that illustrate both graphically and in tabular fashion how the individual sets of models described throughout the remainder of Section 3 fit within the overall system model used to evaluate the system performance.

It is worthwhile to start the introduction to the component models used in the TSPA-SR with the overall performance assessment process. Figure 3.1-1 illustrates the flow of information within the TSPA-SR model as well as the process used in the development of the model and the resulting analyses using the model. Following the FEPs identification and screening, the necessary and sufficient FEPs that need to be included in the TSPA-SR are determined. Section 2.2 has identified this process, the results of which are documented in a series of AMRs that have been summarized in Appendix B.

Once the FEPs that are included in the TSPA have been identified, the FEPs are categorized and lumped into discrete scenario classes. For the TSPA-SR, two scenario classes have been identified, the nominal scenario class and the volcanic scenario class. The volcanic scenario class has been subdivided into two event classes, the volcanic eruption event class and the igneous intrusion event class. In addition to these scenario classes, the TSPA-SR has evaluated a stylized human intrusion scenario, which assumes an inadvertent borehole penetrates the potential repository and the waste package and creates a pathway through the UZ.

For purposes of analysis and presentation of performance assessment results, the model components have been divided into those used to evaluate the nominal performance scenario class and those used to evaluate the disruptive events performance scenario class. The model components developed and implemented to evaluate the robustness of the potential repository system performance given a stylized human intrusion event (i.e., a borehole which is assumed to penetrate the engineered barriers and provide a "short-circuit" through the UZ natural barrier, are treated separately in Section 4.4.1.) The nominal performance scenario class includes those component models which describe the anticipated sequence of processes that are expected to occur during the lifetime of the potential repository system, i.e., those with a probability of occurrence of close to one. The disruptive events performance scenario class includes those component models which describe the sequence of events and processes that, if they occur, could have a significant consequence to public health, but whose probability of occurrence is very small. For purposes of these discussions, high probability events such as seismic activity and climatic change, are included in the nominal performance scenario class. The effects of seismic events on cladding degradation are described in Section 3.5. Low probability events such as volcanism, which has a probability of occurrence of about 0.0002 over the 10,000 year time period of regulatory concern, are considered in the disruptive event performance scenario class.

The proposed DOE regulation 10 CFR Part 963.17 (64 FR 67054 [124754]) requires the consideration of four potentially disruptive processes or events within the context of the total system performance assessment. These four processes and events are (1) volcanism, (2) seismic events, (3) nuclear criticality, and (4) inadvertent human intrusion. Each of these is discussed below.

1. Volcanism–Volcanism is directly included in the volcanic event scenario class. The key information used in the volcanic effects models is summarized in Section 3.10.

2. Seismic Events–With the exception of the effects of seismicity on cladding degradation, seismic induced effects and consequences have been screened out of the TSPA on either probability or consequence criteria (Appendix B).

3. Nuclear Criticality–A detailed postclosure methodology has been developed, as described in detail in the *Disposal Criticality Analysis Methodology Topical Report* (YMP 1998 [104441], Sections 3.1 through 3.7). Evaluations conducted thus far have shown that while there are potentially critical configurations that may occur after 10,000 years, the current waste package designs are expected to meet the remaining criteria. Therefore, this process does not require explicit consideration in the TSPA-SR model.

4. Inadvertent Human Intrusion–A stylized human intrusion scenario has been constructed consistent with guidance available in the proposed EPA standard and NRC regulations. This scenario is described in Section 4.4.

The nominal scenario class includes all relevant processes that must be integrated to yield an assessment of system performance. These processes are shown in schematic form around the TSPA model diagram in Figure 3.1-1. For ease of presentation, the principal component models of the nominal scenario class have been lumped into the following groups:

- UZ Flow (Section 3.2)
- EBS Environments (Section 3.3)
- Waste Package and Drip Shield Degradation (Section 3.4)

- Waste Form Degradation (Section 3.5)
- EBS Transport (Section 3.6)
- UZ Transport (Section 3.7)
- SZ Transport (Section 3.8)
- Biosphere (Section 3.9).

In addition, the principal component models for the volcanic scenario class are summarized in Section 3.10.

The above component models are integrated in the TSPA-SR model to evaluate the system performance of the potential Yucca Mountain repository system. The system performance measures of interest in the TSPA-SR are: (1) the individual dose to an average member of the critical group or the reasonably maximally exposed individual for all likely scenarios (the individual protection standard of proposed 40 CFR Part 197.20 [64 FR 46976 [105065]]), (2) the concentration in the groundwater in a representative volume of water at the point of compliance for the nominal scenario (the groundwater protection standard of proposed 40 CFR Part 197.35 [64 FR 46976 [105065]), and (3) the individual dose to an average member of the critical group for the assumed human intrusion scenario (proposed 40 CFR Part 197.25 [64 FR 46976 [105065]]). In addition, because of the anticipated requirement in the Environmental Impact Statement for evaluating the peak dose over the time period of geologic stability (proposed 40 CFR 197.30 [64 FR 46976 [105065]]), analyses of doses beyond the NRC and EPA 10,000-year period of regulatory concern out to 1,000,000 years have been included in the TSPA-SR.

As a result of the objective to illustrate the confidence in the 10,000-year performance projections and the resiliency of the potential repository system at longer times, analyses of post-closure performance reported on in this document have routinely been conducted to 100,000 years. Although the additional uncertainty associated with these longer term projections has been noted by the EPA and NRC in their draft regulations, these analyses are useful indicators of possible performance for the purposes of evaluating the ability of the Yucca Mountain site to reasonably meet the performance objectives identified by these regulatory agencies. In addition to these 100,000 year analyses, analyses out to the peak dose during the time period of geologic stability (1,000,000 years) have also been conducted.

Categorizing the individual component models required to develop an integrated description of the potential repository system provides a useful means of depicting both how information flows in the analysis as well as the discrete building blocks used to construct the system model. Two different categorizations have been used in previous Yucca Mountain Project documents, namely the attributes of potential repository performance and the process model factors. These same categories are used in the present document to provide a common means of correlating the individual models, analyses, and data sets with other Yucca Mountain related documents.

Figure 3.1-2 illustrates the five attributes of the potential Yucca Mountain repository system. For graphical presentation, each attribute is assigned a distinct shape and color. Figure 3.1-3 illustrates the process model factors that correlate with each of the attributes of the potential repository system performance. Again, each process model factor has a distinct icon, the shape of which corresponds to the shape of the related performance attribute. These icons and shapes allow the reader to navigate through the remaining portions of Section 3 with some understanding of how that particular component fits into the integrated system model. Figures 3.1-4 through 3.1-8 graphically illustrate the relationship between the individual process model factors required to develop the TSPA model and the attributes of the potential repository system performance, as follows:

- Process Model Factors Affecting Water Contacting Waste Package (Figure 3.1-4)
- Process Model Factors Affecting Waste Package Lifetime (Figure 3.1-5)
- Process Model Factors Affecting Radionuclide Mobilization and Release (Figure 3.1-6)
- Process Model Factors Affecting Radionuclide Transport (Figure 3.1-7)
- Process Model Factors Affecting Disruptive Events and Processes (Figure 3.1-8).

In addition to the graphical representation of the interrelationship between the integrated process model factors, Table 3.1-1 presents in tabular form each of the process model factors and the key AMRs that provide the technical basis for the analyses or models used for the input parameters to the TSPA-SR model. This table indicates the source document for the input parameter set to the TSPA-SR model. The underlying models, analyses and data (including the data tracking numbers) that support these AMRs are indicated in Appendix E. The data tracking numbers associated with the output for each run are listed in Appendix G.

Table 3.1-2 presents a tabular correlation of each of the process model factors included in the TSPA-SR with the criteria specified in proposed 10 CFR 963.17 (64 FR 67054 [124754]). This table indicates the completeness of the process model factors used in the TSPA-SR model. This table also indicates which PMR contains the summary and synthesis of the individual process model factor.

The AMRs indicated in Table 3.1-1 and in Appendix E form the technical foundation for the TSPA-SR model. Each of these AMRs has been summarized in a PMR which is discipline-based description of the technical basis for the process models supporting the TSPA for the Site Recommendation Considerations Report. The PMRs cover the following topics:

- Integrated Site Model
- UZ Flow and Transport
- Near Field Environment
- EBS Degradation, Flow and Transport
- Waste Package Degradation
- Waste Form Degradation
- SZ Flow and Transport
- Biosphere
- Disruptive Events.

The scope of these PMRs in summarized in Table 3.1-3. This table indicates the sections of Section 3 which summarize the salient aspects of the individual model and how they are integrated in the TSPA-SR model. Additional details of the integration of the component models in the TSPA-SR model are contained in the *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]).

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Limiting Water Contacting Waste Package	Climate	*Future Climate Analysis* (USGS 2000 [136368])	• Climate states • Timing and sequence	3.2
	Net Infiltration	*Analysis of Infiltration Uncertainty* (CRWMS M&O 2000 [143244])	• Probabilities for different infiltration scenarios	3.2
	UZ Flow	*Abstraction of Flow Fields for RIP (ID:U0125)* (CRWMS M&O 2000 [123913])	• Flow fields for different infiltration scenarios and climate states	3.2
	Coupled Effects on UZ Flow	*Drift-Scale Coupled Processes (DST and THC Seepage) Models* (CRWMS M&O 2000 [142022])	• Flow fields affected by thermal hydrologic	3.2
	Seepage into Emplacement Drifts	*Abstraction of Drift Seepage* (CRWMS M&O 2000 [142004])	• Seepage flux and seepage fraction as a function of percolation flux	3.2
		Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux (CRWMS M&O 2000 [152204])	• Percolation flux: f (multiple locations, waste type, time, climate)	3.2
	Coupled Effects on Seepage	*Abstraction of Drift Seepage* (CRWMS M&O 2000 [142004])	• Seepage flux and seepage fraction as a function of percolation flux	3.2
Prolonging Waste Package Lifetime	In-Drift Physical and Chemical Environments	*Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	• Percolation flux: f (multiple locations, waste type, time, climate) • Temperature and RH on the drip shield surface: f (multiple locations, waste type, time, climate)	3.3
		In-Drift Precipitates/Salts Analysis (CRWMS M&O 2000 [127818])	• pH: f (region, time), response surface • Chloride: f (region, time) • Mass of microbes	3.3
	In-Drift Moisture Distribution	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	• Seepage flux through the drip shield • Fraction of drip shield surface that is wet	3.3

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Prolonging Waste Package Lifetime (Continued)	Drip Shield Degradation and Performance	*Analysis of Mechanisms for Early Waste Package Failure* (CRWMS M&O 2000 [147359])	• Probability of the occurrence of material and manufacturing defect flaws in drip shield • Size of material and manufacturing defect flaws in drip shield	3.4
		Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis (CRWMS M&O 2000 [147641])	• Threshold for general corrosion initiation • General corrosion rate under drip and no-drip conditions • Penetration opening size (or patch size) by general corrosion	3.4
	Drip Shield Degradation and Performance	*Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier* (CRWMS M&O 2000 [146460])	• Threshold for general corrosion initiation	3.4
		WAPDEG Analysis of Waste Package and Drip Shield Degradation 85 CRWMS M&O 2000 [146427])	• Drip shield geometry and thickness • Drip shield failure time history • Number of penetration openings in drip shield by general corrosion, crevice corrosion, SCC, HIC, and other degradation modes	3.4
		In-Drift Precipitates/Salts Analysis (CRWMS M&O 2000 [127818])	pH – f (region, time), response surface Chloride – f (region, time) Mass of microbes	3.4
	Moisture, Temperature, and Chemistry Effects on Waste Package	*Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	Average and maximum temperature on waste package surface – f (waste type, region, time, climate) Temperature and RH on waste package surface – f (multiple locations, waste type, time, climate, infiltration)	3.4
		EBS Radionuclide Transport Abstraction (CRWMS M&O 2000 [129284])	Seepage flux through the drip shield • Fraction of drip shield surface that is wet	3.4

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Prolonging Waste Package Lifetime (Continued)	Waste Package Degradation and Performance	*WAPDEG Analysis of Waste Package and Drip Shield Degradation* (CRWMS M&O 2000 [146427])	• Waste package geometry • Thickness of waste package barriers • Waste package failure time history • Number of penetration openings in waste package by general corrosion, crevice corrosion, SCC, and other degradation modes	3.4
		Aging and Phase Stability of Waste Package Outer Barrier (CRWMS M&O 2000 [147639])	• Kinetics of secondary phase formation in base metal and weld of waste package outer barrier • Threshold secondary phase volume fraction above which corrosion resistance of waste package outer barrier is affected	3.4
		Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier (CRWMS M&O 2000 [146460])	• Threshold RH for general corrosion initiation under drip (after drip shield failure) and no-drip conditions	3.4
		Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis (CRWMS M&O 2000 [147641])	• General corrosion rate under drip (after drip shield failure) and no-drip conditions • Penetration opening size (or patch size) by general corrosion	3.4

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Prolonging Waste Package Lifetime (Continued)	Waste Package Degradation and Performance	*Abstraction of Models for Pitting and Crevice Corrosion of Drip Shield and Waste Package Outer Barrier* (CRWMS M&O 2000 [147648])	• Crevice corrosion initiation threshold of waste package outer barrier • Probability (or area) of crevice formation on waste package outer barrier • Pit density (under crevice) • Pit penetration rate (under crevice) • Pit penetration opening size (under crevice) • Effect of phase stability and aging on initiation and penetration rate of crevice corrosion	3.4
		Abstraction of Models of Stress Corrosion Cracking (SCC) of Drip Shield and Waste Package Outer Barrier and Hydrogen Induced Corrosion of Drip Shield (CRWMS M&O 2000 [135773])	• Stress and stress intensity factor profile in waste package outer barrier • SCC initiation threshold • SCC crack density • SCC crack growth rate • Effect of material and manufacturing defects on SCC initiation and crack growth rate • Effect of rockfall damage on SCC initiation and crack growth rate • SCC crack penetration opening size • Effect of phase stability and aging of waste package outer barrier on SCC initiation and crack growth rate	3.4

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Limiting Rate of Radionuclide Mobilization and Release from the EBS		*Calculation of Probability and Size of Defect Flaws in Waste Package Closure Welds to Support WAPDEG Analysis* (CRWMS M&O 2000 [144551])	• Probability of the occurrence of material and manufacturing defect flaws • Size of material and manufacturing defect flaws	
	Radionuclide Inventory and Distribution in the Potential Repository	*Inventory Abstraction* (CRWMS M&O 2000 [136383])	• Number of packages – Zircaloy-clad fuel – Stainless steel-clad fuel • Inventory per package (actual and adjusted for ingrowth) • Mass fraction	3.5
	In-Package Environments	*In-Package Chemistry Abstraction* (CRWMS M&O 2000 [129287])	• pH – f (region, time) • Total dissolved carbonate (CO32) – f (region, time) • Oxygen fugacity – f (region, time) • Ionic strength – f (region, time) • Fluoride – f (region, time) • CO2 fugacity	3.5
	Cladding Degradation and Performance	*Clad Degradation - Summary and Abstraction* (CRWMS M&O 2000 [147210])	• Fraction of surface area of Zircaloy-clad CSNF exposed as a function of time	3.5
	CSNF Degradation and Performance	*CSNF Waste Form Degradation: Summary Abstraction* (CRWMS M&O 2000 [136060])	• CSNF intrinsic dissolution rate equation, similar to VA • Specific surface area	3.5
	DSNF Degradation and Performance	*DSNF and Other Waste Form Degradation Abstraction* (CRWMS M&O 2000 [144164])	• Dissolution rate equation • Specific surface area	3.5

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Limiting Rate of Radionuclide Mobilization and Release from the EBS (Continued)	Defense HLW Degradation and Performance	*Defense High Level Waste Glass Degradation* (CRWMS M&O 2000 [143420])	• HLW intrinsic dissolution rate equation • Specific surface area	3.5
	Dissolved Radionuclide Concentrations	*Summary of Dissolved Concentration Limits* (CRWMS M&O 2000 [143569])	• Concentration limits (solubilities) for all isotopes included in TSPA	3.5
	Colloid-Associated Radionuclide Concentrations	*Waste Form Colloid-Associated Concentrations Limits: Abstraction and Summary* (CRWMS M&O 2000 [148214])	• Types of waste form colloids • Concentration of colloids • Kd and/or Kc for various colloid types • Fraction of inventory that travels as irreversibly attached onto colloids	3.5
	In-Package Radionuclide Transport	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	• Porosity of corrosion products: f (time) • Saturation of corrosion products: f (time) • Evaporation: f (temperature, RH, composition)	3.6
	EBS (Invert) Degradation and Performance	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	• Invert geometry • Porosity of the invert • Relative permeability of the invert as a function of the saturation • Diffusion coefficient • Volumetric flux through the invert: f (climate, time) • Saturation in the invert after thermal pulse: f (time)	3.6

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Limiting Rate of Radionuclide Mobilization and Release from the EBS (Continued)	EBS (Invert) Degradation and Performance (Continued)	*Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	• Thermally perturbed saturation in the invert: f (waste type, region, time, climate)	3.6
Slow Transport away from the EBS	UZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion)	*Unsaturated Zone and Saturated Zone Transport Properties* (U0100) (CRWMS M&O 2000 [141440])	• Transport parameters — Fracture aperture in different units — Dispersivity of fractures — Dispersivity of matrix • K_d for all isotopes included in TSPA • Matrix diffusion coefficients: f (isotopes, units)	3.7
		Abstraction of Flow Fields for RIP (ID:U0125) (CRWMS M&O 2000 [123913])	• Flow fields for different infiltration scenarios and climate states (FEHM input files [DTN: SN9908T0581699.001[146900]] for the particle tracking model)	
		Particle Tracking Model and Abstraction of Transport Processes (CRWMS M&O 2000 [141418])	• FEHM (data tracking number SN9908T0581699.001) particle tracking model coupled to RIP • Grid nodes for each bin	
		UZ Colloid Transport Model (CRWMS M&O 2000 [122799])	• Kc and/or kinetic colloid parameters for Pu, Am, Th etc. • Colloid filtration factor	
	Coupled Effects on UZ Radionuclide Transport	*Unsaturated Zone and Saturated Zone Transport Properties* (U0100) (CRWMS M&O 2000 [141440])	• N/A Effects of coupled processes on UZ transport have been screened out (see Appendix B)	Appendix B

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Slow Transport away from the EBS (Continued)	SZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion)	*Uncertainty Distribution for Stochastic Parameters* (CRWMS M&O 2000 [147972])	• Flow fields for SZ flux uncertainty • Transport parameters — Effective porosity for all units except the volcanic units — Dispersivity (longitudinal, horizontal transverse, vertical transverse) — Boundary definition of the alluvium • Kd for all isotopes included in TSPA • Matrix porosity • Flowing interval spacing • Effective diffusion coefficient • Flowing interval • Bulk density • Source region definition • Horizontal anisotropy • Kc and/or kinetic parameters for Pu desorption • Colloid filtration factor	3.8
		Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA (CRWMS M&O 2000 [139440])	• Breakthrough curves: f (radionuclide, region) • Input parameters for convolution code • Climate change flux multiplication factor • Capture zones and release locations within each zone	

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Slow Transport away from the EBS (Continued)	Wellhead Dilution	*Groundwater Usage by the Proposed Farming Community* (CRWMS M&O 2000 [144056])	• Annual groundwater use	3.9
	Biosphere Dose Conversion Factors	*Distribution Fitting to the Stochastic BDCF Data* (CRWMS M&O 2000 [144055])	• Biosphere dose conversion factor – f (radionuclide, irrigation time)	3.9
		Abstraction of BDCF Distributions for Irrigation Periods (CRWMS M&O 2000 [144054])	• Biosphere dose conversion factor – f (radionuclide, irrigation time)	
Addressing Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	*Characterize Framework for Igneous Activity at Yucca Mountain, Nevada* (T0015) (CRWMS M&O 2000 [141044])	• Annual probability of igneous intrusion into the waste	3.10
	Characteristics of Volcanic Eruption	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	• Input parameters for ASHPLUME (Jarzemba et al. 1997 [100987]) • Probability that an intrusion into the waste will result in one or more eruptive vents through the waste • Number of vents through the waste for intrusions that result in one or more vents through the waste • Wind Direction Factor	3.10
	Effects of Volcanic Eruption	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	Input parameters for ASHPLUME (Jarzemba et al. 1997 [100987]) • Probability that an intrusion into the waste will result in one or more eruptive vents through the waste • Number of vents through the waste for intrusions that result in one or more vents through the waste • Wind Direction Factor	3.10

Table 3.1-1. Correlation of Process Model Factors with Analysis Model Report Describing Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Process Model Factor	Analysis Model Report	Input Parameters to TSPA-SR	Section of TSPA-SR
Addressing Effects of Potentially Disruptive Processes and Events (Continued)	Atmospheric Transport of Volcanic Eruption	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	• Input parameters for ASHPLUME (Jarzemba et al. 1997 [100987]) • Probability that an intrusion into the waste will result in one or more eruptive vents through the waste • Number of vents through the waste for intrusions that result in one or more vents through the waste • Wind Direction Factor	3.10
	Biosphere Dose Conversion for Volcanic Eruption	*Disruptive Event Biosphere Dose Conversion Factor Analysis* (CRWMS M&O 2000 [143378])	• Biosphere dose conversion factors – f (radionuclide)	3.10
	Probability of Igneous Intrusion	*Characterize Framework for Igneous Activity at Yucca Mountain, Nevada* (T0015) (CRWMS M&O 2000 [141044])	• Annual probability of igneous intrusion into the waste	3.10
	Characteristics of Igneous Intrusion	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	Length and width of dike intersecting drifts • Number of waste packages and drip shields degraded by intrusive event	3.10
	Effects of Igneous Intrusion	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	• Degree of degradation of waste packages, drip shields and cladding due to intrusive event	3.10
	Effects of Seismic Activity	*Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada* (CRWMS M&O 2000 [142321])	• Timing of Seismic event significant enough to degrade engineered barriers	Appendix B 3.5

Table 3.1-2. Correlation of Process Model Factors with Proposed 10 CFR Part 963 Criteria

Key Attributes of System	Process Model Factor	Proposed 10 CFR Part 963 Criteria	PMR	Proposed 10 CFR Part 963 Site Characteristics Criteria 963.17 (a)(1)
Limiting Water Contacting Waste Package	Climate	963.17(a)(2)(i)	UZ[a]	(ii) Hydrologic Properties
	Net Infiltration	963.17(a)(2)(ii)	UZ[a]	(ii) Hydrologic Properties
	UZ Flow	963.17(a)(2)(iii)	UZ[a]	(i) Geologic Properties (ii) Hydrologic Properties (iii) Geophysical Properties
	Coupled Effects on UZ Flow	963.17(a)(3)(i)	UZ[a]	(ii) Hydrologic Properties (iv) Geochemical Properties
	Seepage into Emplacement Drifts	963.17(a)(2)(iv) 963.17(a)(3)	UZ[a]	(i) Geologic Properties (ii) Hydrologic Properties
	Coupled Effects on Seepage	963.17(a)(2)(iv) 963.17(a)(3)	UZ[a]	(ii) Hydrologic Properties (iv) Geochemical Properties
	In-Drift Physical and Chemical Environments	963.17(a)(3) 963.17(a)(3)(i) 963.17(a)(3)(ii)	EBS[b]	(i) Geologic Properties (ii) Hydrologic Properties (iii) Geophysical Properties (iv) Geochemical Properties
Prolonging Waste Package Lifetime	In-Drift Moisture Distribution (Moisture on Drip Shield and Waste Package)	963.17(a)(4)	EBS[b]	(ii) Hydrologic Properties
	Drip Shield Degradation and Performance	963.17(a)(4) 963.17(a)(4)(i)	WP[c]	N/A; See Relevant Engineering Characteristics
	Waste Package Degradation and Performance	963.17(a)(4) 963.17(a)(4)(ii)	WP[c]	N/A

Table 3.1-2. Correlation of Process Model Factors with Proposed 10 CFR Part 963 Criteria (Continued)

Key Attributes of System	Process Model Factor	Proposed 10 CFR Part 963 Criteria	PMR	Proposed 10 CFR Part 963 Site Characteristics Criteria 963.17 (a)(1)
Limiting Rate of Radionuclide Mobilization and Release from the EBS	Radionuclide Inventory and Distribution in the Potential Repository	963.17(a)(5)	WF[d]	N/A
	In-Package Environments	963.17(a)(5)	WF[d]	N/A
	Cladding Degradation and Performance	963.17(a)(5)(i)	WF[d]	N/A
	CSNF Degradation and Performance	963.17(a)(5) 963.17(a)(5)(ii)	WF[d]	N/A
	DSNF Degradation and Performance	963.17(a)(5) 963.17(a)(5)(ii)	WF[d]	N/A
	Defense HLW Degradation and Performance	963.17(a)(5) 963.17(a)(5)(ii)	WF[d]	N/A
	Dissolved Radionuclide Concentrations	963.17(a)(5) 963.17(a)(5)(ii)	WF[d]	N/A
	Colloid-Associated Radionuclide Concentrations	963.17(a)(6)(i)	WF[d]	N/A
	In-Package Radionuclide Transport	963.17(a)(6)(ii)	WF[d]	N/A
	EBS (Invert) Degradation and Performance	963.17(a)(6) 963.17(a)(6)(ii)	EBS[b]	N/A
Slow Transport Away from the EBS	UZ Radionuclide Transport (Advective Pathways, Retardation, Dispersion)	963.17(a)(6) 963.17(a)(6)(i)	UZ[a]	(i) Geologic Properties (ii) Hydrologic Properties (iv) Geochemical Properties
	Coupled Effects on UZ Radionuclide Transport	963.17(a)(7)(ii)	UZ[a]	N/A
	SZ Radionuclide Transport (Advective Pathways, Retardation, Dispersion)	963.17(a)(8) 963.17(a)(8)(i)	SZ[e]	(i) Geologic Properties (ii) Hydrologic Properties(iii) Geophysical Properties (iv) Geochemical Properties

Table 3.1-2. Correlation of Process Model Factors with Proposed 10 CFR Part 963 Criteria (Continued)

Key Attributes of System	Process Model Factor	Proposed 10 CFR Part 963 Criteria	PMR	Proposed 10 CFR Part 963 Site Characteristics Criteria 963.17 (a)(1)
Slow Transport away from the EBS	Wellhead Dilution	963.17(a)(8)(ii)	SZ[e]	N/A
	Biosphere Dose Conversion Factors	963.17(a)(9) 963.17(a)(9)(i) 963.17(a)(9)(ii)	BIO[f]	N/A
Low Mean Annual Dose Even Considering Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	963.17(b)(1)	DE[g]	(i) Geologic Properties (iii) Geophysical Properties
	Characteristics of Volcanic Eruption	963.17(b)(2)	DE[g]	(i) Geologic Properties (iii) Geophysical Properties
	Effects of Volcanic Eruption	963.17(b)(3)	DE[g]	N/A
	Atmospheric Transport of Volcanic Eruption		DE[g]	N/A
	Biosphere Dose Conversion for Volcanic Eruption		BIO[f]	N/A
	Probability of Igneous Intrusion		DE[g]	(i) Geologic Properties (iii) Geophysical Properties
	Characteristics of Igneous Intrusion		DE[g]	(i) Geologic Properties (iii) Geophysical Properties
	Effects of Igneous Intrusion	963.17(b)(4)	DE[g]	N/A

NOTES:
[a] UZ = Unsaturated Zone Flow and Transport Model Process Model Report (CRWMS M&O 2000 [145774])
[b] EBS = Engineered Barrier System Degradation, Flow, and Transport Process Model Report (CRWMS M&O 2000 [145796])
[c] WP = Waste Package Degradation Process Model Report (CRWMS M&O 2000 [138396])
[d] WF = Waste Form Degradation Process Model Report (CRWMS M&O 2000 [138332])
[e] SZ = Saturated Zone Flow and Transport Process Model Report (CRWMS M&O 2000 [145738])
[f] BIO = Biosphere Process Model Report (CRWMS M&O 2000 [151615])
[g] DE = Disruptive Events Process Model Report (CRWMS M&O 2000 [141733])
N/A = not applicable
Proposed 10 CFR Part 963 (64 FR 67054 [124754])

Table 3.1-3. Correlation of Process Model Reports with Applicable Sections of Total System Performance Assessment-Site Recommendation

Process Model Report (PMR)	Scope of PMR	Corresponding Section of TSPA SR
Integrated Site Model[a]	Describes the framework for the geologic properties of the ambient site (e.g., stratigraphy, structural characteristics, geohydrologic rock properties, and mineralogy). The PMR describes how information about the site has been used to characterize the geologic properties of the site.	N/A
UZ Flow and Transport[b]	Describes the processes affecting (1) the amount of water entering the UZ above the potential repository that could contact wastes in the potential repository and (2) the movement of water through the UZ below the potential repository and potential transport of radionuclides in that water. The model describes the spatial and temporal distribution of water flow through the UZ and the spatial and temporal distribution of water concentrations of released radionuclides in the UZ.	3.2 and 3.7
Near-Field Environment[c]	Describes processes important to limiting the amount of water that could contact waste. Processes include the effects of heat from the waste on water flow through the UZ at the emplacement drift wall, seepage, temperature and humidity (thermodynamic environment) near the engineered barriers, and the chemical reactions and products and mechanical interactions in the near-field host rock surrounding the emplacement drifts; water chemistry and gas compositions in the near-field rock are also described.	3.3
EBS Degradation, Flow, and Transport[d]	Describes processes that would lead to degradation of the engineered barriers and affect movement of radionuclides through those barriers. This PMR provides information about the thermal, hydrologic, and geochemical processes acting on engineered barriers relevant to these factors. Water chemistry and gas compositions in the emplacement drifts are also described.	3.3 and 3.6
Waste Package Degradation[e]	Describes processes that could lead to drip shield and waste package degradation (e.g., the corrosion of the waste package materials in the near-field environments).	3.4
Waste Form Degradation[f]	Describes the waste characteristics that limit the rate of release of radionuclides. Processes include HLW waste canister degradation, cladding degradation, and waste form dissolution. The PMR describes the manner in which the waste forms will degrade and how such degradation is expected to affect the release of radionuclides to the immediate environment.	3.5
SZ Flow and Transport[g]	Describes the processes that control the movement of water through the SZ below the potential repository and the distribution of dissolved radionuclides or colloidal particles that might be released from the potential repository and migrate to the SZ. The PMR describes the dilution of radionuclide concentrations during migration in the SZ.	3.8

Table 3.1-3. Correlation of Process Model Reports with Applicable Sections of Total System Performance Assessment-Site Recommendation (Continued)

Process Model Report (PMR)	Scope of PMR	Corresponding Section of TSPA SR
Biosphere[h]	Addresses the characteristics of the biosphere that influence the transport of radionuclides to humans. It includes a description of the lifestyle and habits of individuals who could be exposed to radioactive material at some time during the postclosure performance period. The PMR describes the reference biosphere, associated pathways, and the characteristics of the critical group, including location and behavior representative of current conditions and biosphere transport and uptake parameters.	3.9
Disruptive Events[i]	Summarizes tectonic processes that could disrupt the potential repository system. Other disruptive events, including potential human intrusion and effects of nuclear criticality, are discussed elsewhere. This PMR focuses on the consequences of volcanic and seismic events that could affect the potential repository system. The consequence analyses rely on inputs from the probabilistic volcanic hazard analysis and probabilistic seismic hazard analysis to describe the frequency of disruptive events. Effects of igneous activity on drifts, waste packages, and waste forms are addressed. Modes of radionuclide release resulting from igneous events are also characterized. Effects of seismic activity on the EBS and on waste package degradation are summarized. The potential effects of volcanic and seismic activity on the hydrologic system are discussed.	3.10

NOTES:
[a] CRWMS M&O 2000 [146988]
[b] CRWMS M&O 2000 [145774]
[c] CRWMS M&O 2000 [153178]
[d] CRWMS M&O 2000 [145796]
[e] CRWMS M&O 2000 [138396]
[f] CRWMS M&O 2000 [138332]
[g] CRWMS M&O 2000 [145738]
[h] CRWMS M&O 2000 [151615]
[i] CRWMS M&O 2000 [141733]

A final means of correlating the process model factors included in the TSPA-SR is indicated in Table 3.1-4 where the NRC's KTIs are correlated to the process model factors used in the TSPA-SR.

In summary, this introduction provides a roadmap for the technical discussions which follow in the remainder of Section 3. These technical presentations provide insights into the key inputs to the TSPA model and provide a foundation for understanding the system and subsystem performance results presented in Section 4 and the sensitivity analyses presented in Section 5.

3.2 UNSATURATED ZONE FLOW

UZ flow at Yucca Mountain potentially plays an important role in potential repository performance. Water seeping into drifts and dripping onto waste packages can accelerate radionuclide mobilization and release. In addition, the amount of water flow in fractures from the potential repository to the water table is an important factor in determining the transport time of radionuclides to the biosphere.

Four distinct components of unsaturated flow are considered in the TSPA for the Site Recommendation: climate, infiltration, mountain-scale flow of water, and seepage into repository emplacement drifts. These components include processes at several scales. At the global scale is climate, including changes in solar heating caused by changes in the Earth's orbit and inclination, and formation of huge ice sheets during glacial periods. There are also important regional-scale climate effects, such as the rain shadow caused by the Sierra Nevada mountain range and the proximity of the polar jet stream, that make climate variations at Yucca Mountain different from the global average. Infiltration and flow through the mountain, or mountain-scale flow, are modeled at the scale of the site. The site-scale models include effects of surface topography and subsurface hydrogeologic layering. The drift scale is the scale of an emplacement drift, approximately 5.5 m (18 ft) in diameter. At the drift scale, the seepage model evaluates the interaction of percolating water with an emplacement drift and the amount of water that seeps into the drift. Processes at the scale of individual fractures, in particular those affecting fracture-matrix coupling, or water flow between fractures and the porous rock matrix, are important to both seepage and mountain-scale flow. Fracture-scale processes are not modeled explicitly but are represented by parameters in the continuum-flow models. Some of the important processes for UZ flow are pictured in Figure 3.2-1.

As discussed in Sections 1.6.1 and 2.1, an important step in conducting a TSPA is a systematic consideration of features, events, and processes (FEPs) that are relevant to the potential repository system (natural and engineered). An extensive list of FEPs has been developed for Yucca Mountain. The FEPs that are concerned with UZ flow and transport are discussed in detail in *Features, Events, and Processes in UZ Flow and Transport* (CRWMS M&O 2000 [142945]). That report discusses 81 primary FEPs, including whether or not they are included in the TSPA models and, if not, the justification for excluding them (an argument that either the probability or the consequence is small). Of the 81 primary UZ flow and transport FEPs, 22 are fully included in the TSPA. Four of those are strictly related to radionuclide transport, which leaves 18 fully included primary FEPs related to UZ flow. These include such standard processes as "global climate change" (Section 3.2.1) and "fracture flow in the UZ" (Section 3.2.3).

Table 3.1-4. Correlation of Process Model Factors with U.S. Nuclear Regulatory Commission Key Technical Issues

Key Attributes of System	Process Model Factor	NRC Key Technical Issue
Limiting Water Contacting Waste Package	Climate	Unsaturated and Saturated Flow under Isothermal Conditions
	Net Infiltration	
	UZ Flow	
	Coupled Effects on UZ Flow	Thermal Effects on Flow
	Coupled Effects on Seepage	
	Seepage into Emplacement Drifts	Repository Design and Thermomechanical Effects
	In-Drift Physical and Chemical Environments (Environments on Drip Shield and Waste Package)	Evolution of the Near-Field Environment
Prolonging Waste Package Lifetime	In-Drift Moisture Distribution (Moisture on Drip Shield and Waste Package)	
	Drip Shield Degradation and Performance	Container Life and Source Term
	Waste Package Degradation and Performance	
Limiting Rate of Radionuclide Mobilization and Release from the EBS	Radionuclide Inventory and Distribution in Potential repository	
	In-Package Environments	
	Cladding Degradation and Performance	
	CSNF Degradation and Performance	
	DSNF Degradation and Performance	
	Defense HLW Degradation and Performance	
	Dissolved Radionuclide Concentrations	
	Colloid-Associated Radionuclide Concentrations	
	In-Package Radionuclide Transport	
	EBS (Invert) Degradation and Performance	

Table 3.1-4. Correlation of Process Model Factors with U.S. Nuclear Regulatory Commission Key Technical Issues (Continued)

Key Attributes of System	Process Model Factor	NRC Key Technical Issue
Slow Transport away from the EBS	UZ Radionuclide Transport (Advective Pathways, Retardation, Dispersion)	Radionuclide Transport
	Coupled Effects on UZ Radionuclide Transport	
	SZ Radionuclide Transport (Advective Pathways, Retardation, Dispersion)	
	Wellhead Dilution	
	Biosphere Dose Conversion Factors	
Low Mean Annual Dose Even Considering Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	Igneous Activity
	Characteristics of Volcanic Eruption	
	Effects of Volcanic Eruption	
	Atmospheric Transport of Volcanic Eruption	
	Biosphere Dose Conversion for Volcanic Eruption	
	Probability of Igneous Intrusion	
	Characteristics of Igneous Intrusion	
	Effects of Igneous Intrusion	
	Effects of Seismic Activity	Structural Deformation and Seismicity

In addition to those FEPs fully included, many FEPs are partially included, for example "climate modification increases recharge." This FEP is included in the TSPA (Section 3.2.2). However, one of the secondary FEPs under this primary FEP relates to the possible formation of new perched water bodies below the potential repository because of the increased water flow, and this aspect is not included in the TSPA model on the basis of low consequence (CRWMS M&O 2000 [142945], Section 6.3.4). Lastly, a number of FEPs are excluded entirely from the TSPA, including ones relating to erosion and subsidence (CRWMS M&O 2000 [142945], Section 6.4), human influences on climate (CRWMS M&O 2000 [142945], Section 6.5), and gas generation (CRWMS M&O 2000 [142945], Section 6.6). A complete list of the primary FEPs and their status (whether included or excluded) is given in Appendix B.

The following sections discuss, in turn, climate, infiltration, mountain-scale flow, and seepage into drifts. For each one, the important processes and assumptions, the implementation for use in TSPA simulations, the treatment of uncertainty and variability, and some results and interpretations are given. The relationships of these submodels with each other and with other TSPA components are diagrammed in Figure 3.2-2. More detailed information on UZ flow, including full description and justification of the models and data as well as potential alternative modeling approaches where appropriate, can be found in the *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774]) and the associated AMRs.

3.2.1 Climate

3.2.1.1 Features, Processes, and Conceptual Model

Climate refers to the meteorological conditions that characteristically prevail in a particular region. Climate conditions at Yucca Mountain must be known to determine the hydrology within and around Yucca Mountain. In particular, temperature and precipitation are important inputs to the infiltration model.

The climate in the Yucca Mountain region is presently warm and semiarid. However, past climates have exhibited a great deal of variability, alternating between glacial and interglacial periods. During glacial climates, glaciers did not reach Yucca Mountain, but temperatures were lower and precipitation was higher, resulting in greater effective moisture relative to today.

In the conceptual model for future climate at Yucca Mountain, the climate is approximated by a sequence of discrete climate states (see Figure 3.2-3). The bases for this approach are as follows (CRWMS M&O 2000 [145774], Section 3.5.1.2):

- Climate is cyclical, with an approximate period of 400,000 years, and this pattern is expected to continue into the future. (Climate is not purely cyclical, however, but has a chaotic component superimposed on the basic cyclical behavior.)

- The timing of past (and future) climates is related to the precession of the Earth's rotational axis (i.e., how the Earth wobbles like a spinning top) and changes in the Earth's orbital eccentricity (i.e., deviation of the orbit from circularity).

- The characteristics of past climates and the sequence of those climates are related. Thus, the analysis can focus on a particular climate sequence and need not take the conservative approach of using the climates that generate the highest infiltration.

- Long-term, Earth-based, climate-forcing processes, such as tectonics, have remained relatively constant over the past 500,000 years or so and should remain so for the period of interest for performance assessment. This is important to climate forecasting because such forcing processes can change climate in noncyclic ways, invalidating the other bases listed above. The effects of such noncyclic forcing processes make climate projections less reliable over longer time periods.

3.2.1.2 Implementation in the Total System Performance Assessment

The timing and properties of the climate states were derived primarily from the microfossil record of cores drilled at Owens Lake, California (roughly 160 km, or 100 miles, to the west of Yucca Mountain). The timing of past climates is based on estimates of the rate at which sediment accumulated in the lake bed over time and comparison with the chronology that has been published for calcite deposition at Devils Hole, Nevada (about 50 km, or 30 miles, southeast of Yucca Mountain). The properties of the past climates (e.g., precipitation and temperature) are based on detailed examination of the Owens Lake cores, in particular for the abundances of various species of ostracode (small freshwater crustaceans) through time. The environmental tolerances of ostracode species provide a way to estimate the relative water temperature and salinity, thus providing a proxy for climate variations through time (CRWMS M&O 2000 [145774], Section 3.5.1.3).

The 10,000-year regulatory period has been divided into three climate regimes: modern (present-day), monsoon, and glacial-transition (Figure 3.2-3). Present-day analogs were chosen to represent the future climates; that is, locations that currently have climatic conditions analogous to those deduced for the past climates. Note that the climate discussions in the Process Model Report (CRWMS M&O 2000 [145774], Section 3.5) and in the associated climate AMR (USGS 2000 [136368]) are limited to 10,000 years in the future. Because longer periods of time were not analyzed, the TSPA base case simply extends the glacial-transition climate when simulating periods longer than 10,000 years.

The connections between climate and other TSPA model components are shown in Figure 3.2-4. The climate-change times are the only climate information used directly by the TSPA model. The other information is used indirectly, through its effects on infiltration. Additional climate effects are discussed in Section 3.2.3 (climate-induced change in the water table) and Section 3.8.1.1 (climate-induced change in saturated-zone percolation flux).

3.2.1.3 Treatment of Uncertainty and Variability

Since climate is a large-scale phenomenon, no spatial variability of climate is included in the Yucca Mountain TSPA model. Temporal variability is included as a sequence of climate states, as described above. To capture uncertainty, upper-bound and lower-bound climate analogs were chosen for each climate state, with the upper bound representing wetter conditions and the lower bound representing drier conditions.

The TSPA base case does not include uncertainty in climate durations. However, many uncertainties do exist that may affect the timing of the three climate states. These include uncertainties in the Owens Lake sediment-accumulation rate, uncertainties in the methods used to identify what time period in the past is most like the future period being modeled, the existence of a chaotic component of climate, and the possibility that human activity may change the cyclical patterns. A quantitative estimate has been made of the first uncertainty (the Owens Lake sediment-accumulation rate), leading to a range of 400 to 600 years for the duration of the present-day climate and 900 to 1,400 years for the duration of the monsoon climate, with the glacial-transition climate extending for the balance of the model period in each case (CRWMS M&O 2000 [145774], Section 3.5.1.5). There is no simple or objective way of assessing the other uncertainties quantitatively. However, the uncertainty in the time of climate change can be neglected in the TSPA because it has little impact on the dose results. In the current TSPA model, the waste packages are so robust that there are no failures for over 10,000 years, which makes climate-change uncertainties of a few hundred years, or even a few thousand years, insignificant. In the TSPA for the VA, effects of uncertainty in climate-change time were investigated and found to be unimportant (DOE 1998 [100550], Volume 3, Section 5.1.1).

3.2.1.4 Results and Interpretation

The climate analogs and durations for the TSPA base case are summarized in Table 3.2-1. Multiple analog sites were chosen to represent the future-climate bounds in order to minimize the influence of local meteorological phenomena (CRWMS M&O 2000 [145774], Section 3.5.1.4).

Table 3.2-1. Durations and Analog Sites for Climate States

Climate State	Duration	Representative Meteorological Stations
Present-Day Climate	600 years	Yucca Mountain and vicinity
Monsoon Climate	1,400 years	Upper Bound: Nogales, Arizona Hobbs, New Mexico
		Lower Bound: Yucca Mountain and vicinity
Glacial-Transition Climate	Remainder	Upper Bound: Spokane, Washington Rosalia, Washington St. John, Washington
		Lower Bound: Beowawe, Nevada Delta, Utah

Source: CRWMS M&O 2000 [145774], Table 3.5-1

As indicated in Figure 3.2-3 and Table 3.2-1, the modeled climate consists of three periods: 600 years with present-day climate, followed by 1,400 years of a monsoon climate, and then a glacial-transition climate for the remainder of the model period. The monsoon climate is somewhat warmer than present and with a potential to be significantly wetter; in particular, having more summer rain. The glacial-transition climate is cooler than present-day and possibly significantly wetter, with relatively cool, dry summers and cool, wet winters (CRWMS M&O 2000 [145774], Section 3.5.1.4). Each climate state has a range of possible behaviors, represented by bounding analog meteorological sites. Note, however, that the upper-bound climate analogs still fall under the definition of semiarid. Results for the range of net infiltration for each climate state are given in Section 3.2.2.4.

3.2.2 Infiltration

3.2.2.1 Features, Processes, and Conceptual Model

Net infiltration is the penetration of water through the ground surface to a depth where it can no longer be withdrawn by evaporation or transpiration by plants. Infiltration occurs once water has entered bedrock or has penetrated below the root zone in soil. The conceptual model used for infiltration calculations is based on evidence from field studies at Yucca Mountain, combined with established concepts in soil physics and hydrology. The overall framework of the conceptual model is provided by the hydrologic cycle, including processes on the surface and just below the surface that affect net infiltration. Important infiltration processes are illustrated in Figure 3.2-5.

The main components of net infiltration are precipitation, evapotranspiration (evaporation plus transpiration), and surface-water runoff and run-on. These components are incorporated into a watershed-scale, volume-balanced model using a snowpack submodel, an evaporation and net radiation submodel, one-dimensional (vertical) root-zone infiltration submodels, and a two-dimensional surface-water, flow-routing submodel (CRWMS M&O 2000 [145774], Section 3.5.2.4). Precipitation rate is provided by the analog meteorological stations discussed in Section 3.2.1.4 and is spatially distributed based on an empirical precipitation-elevation relationship. Evapotranspiration is the combined process of evaporation and transpiration, which is the removal of moisture from soil by plants. The potential evapotranspiration is determined by an energy balance and is primarily dependent on net radiation, air temperature, ground heat flux, the saturation-specific humidity curve, and wind. Temperature is also spatially distributed for the model by using an empirical temperature-elevation relationship. Net infiltration is modeled through soil layers of the root zone only. The change in root-zone water content is calculated as a daily change in water content, using a simple water-balance approach. The amount of daily net infiltration is limited by the bulk saturated hydraulic conductivity of the soil and storm duration. Water that exceeds the infiltration capacity of a soil column is routed to lower elevation nodes (runoff) for subsequent infiltration (CRWMS M&O 2000 [145774], Section 3.5.2.4).

Considerable site information is available, including water-content profiles from 98 neutron moisture probes on Yucca Mountain from 1984 to 1995 (CRWMS M&O 2000 [145774], Section 3.5.2.3) and daily mean discharge for five stream gauges in operation at Yucca Mountain during 1994 to 1995 (USGS 2000 [123650], Section 6.8).

3.2.2.2 Implementation in the Total System Performance Assessment

Spatially distributed net infiltration rates were determined for each of the three climate states using the infiltration process model. The model covers an area of 124 km^2 (48 square miles) around Yucca Mountain, using a regular grid with 30-m (98-ft) spacing. The inputs to the model include daily precipitation and temperature; ground-surface elevation and geometry; parameters to describe surface water flow, soil type and depth; and bedrock type (USGS 2000 [123650], Section 6.3.3).

For the upper-bound and lower-bound future climate states, daily precipitation and temperature data from the analog meteorological stations (Table 3.2-1) were used directly. For present-day climate, data from Yucca Mountain and nearby meteorological stations were used. The present-day meteorological data were supplemented with a 100-year stochastic precipitation model because most of the records in the Yucca Mountain region go back less than 50 years (USGS 2000 [123650], Section 6.9.2).

In addition to using different meteorologic inputs, future-climate conditions are also represented by using different root-zone parameters to account for increases in vegetation density and changes in vegetation type for wetter and colder future climates (USGS 2000 [123650], Section 6.9.4).

The connections between infiltration and other TSPA model components are shown in Figure 3.2-6. Infiltration is only used by the TSPA model indirectly through its effects on mountain-scale flow and thermal hydrology. The probabilities for the three infiltration cases (Section 3.2.2.3) are used directly by the TSPA model.

3.2.2.3 Treatment of Uncertainty and Variability

In the infiltration model, spatial variability is included by use of a two-dimensional model that incorporates the appropriate surface topography and geology. Temporal variability is included by using different infiltration simulations for different climate states; precipitation, temperature, and vegetation are varied with climate.

In order to represent infiltration uncertainty in TSPA simulations, three infiltration maps were generated for each climate state. They are termed the low-, medium-, and high-infiltration cases. For present-day climate, the medium-infiltration case was developed using the best estimates for precipitation, temperature, etc.; the high-infiltration case was developed by including precipitation data from one of the wetter nearby meteorological stations (on Rainier Mesa); and the low-infiltration case was developed by selectively choosing low infiltrations for each spatial location from a set of infiltration simulations. For future climates, the low-infiltration cases were simulated using the lower-bound climate analogs, the high-infiltration cases were simulated using the upper-bound climate analogs, and the medium-infiltration cases were developed from the low- and high-infiltration maps by averaging them at each spatial location. Where multiple analog sites are indicated in Table 3.2-1, multiple infiltration simulations were performed, and the resulting infiltration values were averaged at each spatial location. Note that the low-infiltration case for the monsoon climate is defined to be the same as the medium-infiltration case for the present-day climate, due to the selection of present-day Yucca Mountain as the lower-bound analog for the monsoon climate (USGS 2000 [123650], Section 6.9.3).

In order to include the three infiltration cases for each climate state into TSPA simulations, it is necessary to estimate their relative probability or likelihood. The probabilities were derived by means of a detailed analysis of infiltration uncertainty using the Monte Carlo method. The uncertainty analysis focused on 12 key input parameters, for which uncertainty distributions were developed. The key parameters include precipitation, bedrock and soil hydrologic properties, and potential evapotranspiration. The infiltration model was run 100 times, with each realization having different sampled values of the 12 key parameters. The glacial-transition climate was used for the analysis, because that climate state is in effect most of the time, and it is in effect at later times when radionuclide releases are more likely. Probabilities for the three infiltration cases were assigned in such a way as to make the log mean and standard deviation of the three-point discrete distribution (i.e., the distribution consisting of the three infiltration cases with defined probabilities) equal to the log mean and standard deviation of the distribution from the Monte Carlo simulation. Log values were used in deriving the weighting factors because the distribution has a more normal shape in log space than in linear space (CRWMS M&O 2000 [145774], Sections 3.5.2.6 and 3.5.3.2).

3.2.2.4 Results and Interpretation

Results of the infiltration modeling are summarized in Table 3.2-2. The table shows computed average net infiltration over the repository area for the three infiltration cases and three climate states. In addition, the final column shows the estimated probabilities for the three infiltration cases. The infiltration numbers in Table 3.2-2 are plotted as time histories of repository-average net infiltration for the three infiltration cases in Figure 3.2-7. The results show the monsoon climate to have higher net infiltration than the present-day climate, which is expected because of the predicted higher precipitation rate. In addition, the glacial-transition climate is shown to have higher net infiltration than the monsoon climate, except for the low-infiltration case. The increase in net infiltration is a result of the colder temperatures, which reduce the amount of evapotranspiration. The glacial-transition low-infiltration case has lower infiltration than the monsoon low-infiltration case because of a more uniform seasonal distribution of precipitation. The average intensity and frequency of precipitation events for this case are not sufficient to overcome evapotranspiration from the root zone (USGS 2000 [123650], Section 6.11.3).

Table 3.2-2. Average Net Infiltrations and Probabilities for the Infiltration Cases

Infiltration Case	Present-Day Climate	Monsoon Climate	Glacial-Transition Climate	Probability (All Climate States)
Low infiltration	0.4 mm/yr	4.7 mm/yr	2.2 mm/yr	17%
Medium infiltration	4.7 mm/yr	13 mm/yr	20 mm/yr	48%
High infiltration	12 mm/yr	20 mm/yr	37 mm/yr	35%

Source: USGS 2000 [123650], Tables 6-10, 6-14, and 6-19; CRWMS M&O 2000 [145774], Section 3.5.3.2
NOTE: 25.4 mm = 1 in.

Simulated net infiltration at the surface is shown as the left-hand map in Figure 3.2-8. Shown are results for the medium-infiltration case, present-day climate. In order to have consistency with the other two flux maps in the figure, the infiltration is illustrated for the domain of the site-scale UZ-flow model (Section 3.2.3.2). For the figure, net infiltration has been averaged somewhat for use by the UZ-flow model, and the numerical mesh for the UZ-flow model is superimposed for reference; the potential repository is located in the middle of the region, where there is a relatively uniform rectangular grid. The highest net infiltrations occur along the crest of the mountain, where less alluvial cover allows more water to penetrate into the bedrock without being evaporated, and in washes, where there is significant runoff and run-on.

3.2.3 Mountain-Scale Flow

3.2.3.1 Features, Processes, and Conceptual Model

UZ flow refers to the percolation of groundwater through rocks above the water table. UZ flow varies with the rock strata through which the water flows. As depicted in Figure 3.2-9, the UZ in Yucca Mountain is composed of alternating layers of welded and nonwelded tuffs. The terms welded and nonwelded refer to the degree of consolidation of the rock when it was formed from volcanic ash millions of years ago. Welded tuff is hard, dense rock but is typically very fractured. The welded-tuff rock matrix is relatively impermeable, but the total permeability is high because of the fractures. Nonwelded tuff is softer and typically less fractured. Its matrix permeability is much higher than that of welded tuff matrix, but its total permeability is typically lower. A special case is zeolitic nonwelded tuff, which was altered during the original cooling in such a way that its matrix permeability is very low (similar to the matrix permeability of welded tuff). Zeolitic tuff may also have little fracturing, so that its total permeability is very low. Perched-water zones (localized saturated regions above the water table) have been observed on the zeolitic tuff in several boreholes, well below the potential repository location (CRWMS M&O 2000 [145774], Section 3.3.8).

Fractures play an important role in UZ flow. This role is especially important when considering radionuclide transport from a nuclear-waste repository. Flow in fractures is typically much faster than flow in the porous matrix, potentially leading to fast transport to the water table. The rock-matrix flow capacity for the welded tuff (typically on the order of 1 mm/yr, or 0.04 in./yr, depending on the hydrogeologic unit and the infiltration case) is less than most estimates of average infiltration (see Table 3.2-2), implying that much of the flow is moving through fractures, especially for the wetter climate states. Field observations and modeling results indicate limited interaction between fracture and matrix flows in the welded tuff (CRWMS M&O 2000 [145774], Section 3.3.4).

The mountain-scale UZ flow model uses a dual-permeability conceptual flow model that captures the effects of fast flow paths and allows for weak fracture-matrix coupling. Flow is modeled in two interacting continua (fracture and matrix), with each continuum assigned its own spatially variable hydrologic properties, such as permeability and porosity. In general, the fractures are modeled as a highly permeable continuum having low porosity, while the matrix is modeled as a much less permeable continuum having higher porosity. Fracture-matrix interaction is represented with an active-fracture model, in which only a portion of the fractures is actively flowing under unsaturated conditions (CRWMS M&O 2000 [145774], Section 3.3.4). Major faults are included in the model explicitly. In fault zones, fracture density and permeability are higher than in the rest of the model, which enables them to act as preferential flow paths in parts of the model.

The data used to develop the mountain-scale calibrated flow fields for the UZ come from surface-based drillholes and from the Exploratory Studies Facility (ESF), which is an 8-km-long (5-mile-long) tunnel through Yucca Mountain. These data include rock-matrix saturations, water potentials (the ability for rock to hold water), temperatures, perched-water locations and amounts, chemical composition and isotopic abundances of groundwater and mineral deposits, air permeability and air-pressure measurements, rock types, fault locations and offsets, fracture density, and matrix permeability and saturation/desaturation parameters. More detailed information on data and the calibration procedure can be found in *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774], Section 3.6).

Wetter conditions in the past suggest that more groundwater flowed beneath Yucca Mountain than flows under modern conditions. Several studies have shown that the water table has risen at most 115 m (380 ft) in the past (CRWMS M&O 2000 [123913], Section 6.2). For monsoon and glacial-transition climates in the TSPA simulations, the water table is conservatively set to an elevation of 850 m (2790 ft) above sea level, which is approximately 120 m (390 ft) higher than the present-day water table under the potential repository (CRWMS M&O 2000 [145774], Section 3.7.5.2). Note that the potential repository is planned to be over 300 m (980 ft) above the present water table, so the potential repository would still be well above the water table, even with a rise of 120 m (390 ft).

The mountain-scale UZ flow model is a steady-state, isothermal model. Thermal effects can be neglected because flow is strongly perturbed by heat only near the emplacement drifts and at early times (CRWMS M&O 2000 [145774], Sections 1.2.3, 3.3.12, 3.10, 3.12). There is additional discussion of thermal effects on flow in Section 3.3.3, though the emphasis there is on flow near the emplacement drifts. Short-term episodic transients can be neglected because of the buffering effect of the Paintbrush nonwelded hydrogeologic unit above the potential repository (CRWMS M&O 2000 [145774], Sections 1.2.3, 3.3.6, 3.3.7, 3.7.3.1). Long-term transients due to climate change are expected to be important to calculations of potential-repository performance, and those are included in the TSPA by approximating mountain-scale UZ flow with a sequence of steady states. With this approximation, the UZ flow fields can be calculated with the steady-state model and then the UZ transport model can simply switch from one flow field to another at climate-change times (Section 3.7.2).

3.2.3.2 Implementation in the Total System Performance Assessment

The three-dimensional, site-scale flow model for the UZ (CRWMS M&O 2000 [145774], Section 3.7) was used in calculating mountain-scale UZ flow for this TSPA. Three-dimensional modeling is needed primarily because of flow characteristics below the potential repository, where significant lateral flow is believed to occur as a result of low-permeability zeolitic layers and perched water. The model uses the dual-permeability conceptual model, as discussed in Section 3.2.3.1. The direct use of the three-dimensional, mountain-scale flow model eliminates the need to test simplified abstractions against more complex process models. The models were tested directly against site data as part of the calibration procedure (CRWMS M&O 2000 [145774], Sections 3.6 to 3.8). The following paragraphs summarize the most important features of the mountain-scale UZ flow model.

The mountain-scale flow model is based on site data that include core measurements of porosity, permeability, saturation, and moisture potential; perched-water observations; fracture-frequency data from the ESF; air-pressure monitoring and air-permeability tests; and geochemical data. These data were used to estimate hydrologic parameters through direct calculations and calibration methods (CRWMS M&O 2000 [145774], Section 3.6). In addition, a geologic-framework model was used to define the proper stratigraphy for the UZ flow model (CRWMS M&O 2000 [145774], Section 3.2; CRWMS M&O 2000 [146988]).

The numerical discretization of the mountain-scale flow model is illustrated in Figure 3.2-10. Figure 3.2-10a shows an east-west cross section through the integrated site model. The cross section has been chosen to intersect three drill holes, named SD-6, SD-12, and UZ#16. The stratigraphy and important hydrogeologic properties are known at the locations of the drill holes; at other locations, they are inferred. The stratigraphic division into hydrogeologic units is shown, including offsets for four major faults. The nomenclature for identifying hydrogeologic units will not be explained here, but it can be found in *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774], Section 3.2.2). Of particular interest are the host units for the potential repository, which are the Topopah Spring middle nonlithophysal (Tptpmn), lower lithophysal (Tptpll), and lower nonlithophysal (Tptpln) units. A higher-level division into TCw (Tiva Canyon welded), PTn (Paintbrush nonwelded), TSw (Topopah Spring welded), and CHn (Calico Hills nonwelded) is indicated at the right of the cross section. These divisions are important hydrologically, with the welded units having primarily fracture flow, as discussed in Section 3.2.3.1, and the nonwelded units having primarily matrix flow. Figure 3.2-10b shows the same cross section as it is represented in the mountain-scale flow model. In the model, the stratigraphy is discretized and simplified to a certain extent (e.g., by making the Ghost Dance fault vertical rather than slightly inclined). The second cross section shows the designators used for the hydrogeologic units within the UZ flow model (e.g., the repository host units are tsw34 [middle nonlithophysal], tsw35 [lower lithophysal], and tsw36 [lower nonlithophysal]). Figure 3.2-10c shows the model discretization as seen from above.

UZ flow calibration was done only for the present-day climate, because there is not sufficient information about paleohydrology to be able to calibrate flow for monsoon and glacial-transition climates. Thus, the flow for future climates is simulated using the same calibrated hydrologic properties and only changing the infiltration boundary condition at the surface. The three calibrated property sets (low-infiltration, medium-infiltration, and high-infiltration) times three climate states lead to a total of nine mountain-scale UZ flow fields in the TSPA.

The connections between mountain-scale UZ flow and other TSPA model components are shown in Figure 3.2-11. The nine UZ flow fields are used directly by the TSPA model in simulating UZ radionuclide transport. In addition, UZ hydrologic properties and stratigraphy are used by the seepage, thermal hydrology, and UZ transport models.

3.2.3.3 Treatment of Uncertainty and Variability

In the UZ flow model, spatial variability is included by use of a three-dimensional model that incorporates the appropriate geometry, geology, and hydrostratigraphy. Temporal variability is included by using different UZ flow simulations for different climate states. The primary change in going from one climate to another is in the use of a different infiltration map for the upper boundary condition. In addition, the water table elevation is higher for future climates.

The uncertainty regarding the appropriate level of net infiltration, discussed in Section 3.2.2.3, was carried forward through the mountain-scale flow model. Sensitivity analyses have shown that infiltration is the quantity that has the greatest impact on UZ flow and transport and that other uncertainties have little impact on repository performance (e.g., DOE 1998 [100550], Volume 3, Section 5.1.3). Although there is a considerable amount of data for calibrating the mountain-scale flow model, the hydrologic properties are not fully constrained, and it is possible to develop calibrated mountain-scale flow models for all three infiltration cases (CRWMS M&O 2000 [145774], Section 3.6). Additional calibrated models were developed with a different treatment of the flow in the vicinity of the perched water between the potential repository level and the water table (CRWMS M&O 2000 [145774], Section 3.7.3.3). However, the alternative perched-water treatment did not lead to greatly different radionuclide transport times from the potential repository to the water table; consequently, only the more conservative of the two perched-water models (perched-water model #1) was used for TSPA simulations (CRWMS M&O 2000 [145774], Section 3.7.5.3).

3.2.3.4 Results and Interpretation

Nine flow fields are used, and the average infiltrations in Table 3.2-2 also represent the average flux over the repository at the top boundary of the flow model. As discussed below, the percolation flux at the potential repository is similar to the infiltration imposed at the surface, but the percolation flux at the water table is distributed quite differently because of significant lateral diversion between the potential repository and the water table. The probabilities for the three infiltration cases apply also to the UZ flow fields derived from those infiltration cases. Because the flow fields represent a large quantity of three-dimensional information, they are not easily summarized. In addition to the short discussion in the remainder of this section, additional description and visualization of the flow fields can be found in Section 3.7.4 of this report (UZ transport), *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774], Section 3.7); *UZ Flow Models and Submodels* (CRWMS M&O 2000 [122797], Sections 6.6 to 6.7); and *Analysis of Base-Case Particle Tracking Results of the Base-Case Flow Fields* (CRWMS M&O 2000 [134732], Section 6.2).

Figure 3.2-8 shows how the mountain-scale flow changes as it moves downward through the mountain. The figure shows the net infiltration flux that enters the top of the model at the ground surface (left), the percolation flux at the potential repository level (center), and the percolation flux that leaves the bottom of the model at the water table (right). The case pictured is medium infiltration for present-day climate. The model numerical mesh is superimposed for reference; the potential repository is located in the middle of the region, where there is a relatively uniform rectangular grid.

Comparison of the three flux maps in Figure 3.2-8 shows that the simulated distribution of percolation flux at the potential repository level is similar to the distribution of net infiltration at the surface, but the distribution of percolation flux at the water table is quite different from the other two, indicating significant nonvertical flow between the potential repository and the water table. The high-infiltration band along the crest of the mountain in the left-hand map is somewhat spread out by the time it arrives at the potential repository (center map), and a portion of the flow has been diverted down the Solitario Canyon fault. (For location of the Solitario Canyon fault, see Figure 3.2-10c.) Similarly, there is significant flow down the other faults as well. At the water table (right-hand map), much of the water is flowing down the faults, more so in the north than in the south. The fraction of total flow that is in the faults over the entire model domain increases from 4 percent at the ground surface to 15 percent at the potential repository level to 35 percent at the water table (CRWMS M&O 2000 [145774], Table 3.7-3). The diversion of water into flow down faults is much less strong in the southern half of the model domain. There, the fraction of flow in faults increases from 4 percent at the ground surface to 20 percent at the potential repository level and then decreases slightly to 16 percent at the water table (CRWMS M&O 2000 [145774], Table 3.7-3). The difference in behavior between the northern and southern parts of the model domain occurs because the Calico Hills nonwelded tuff is largely zeolitized in the north, resulting in a low-permeability zone and extensive perched water, which causes strong lateral diversion to more permeable fault zones in the simulation (CRWMS M&O 2000 [145774], Figure 3.7-9).

3.2.4 Seepage into Drifts

3.2.4.1 Features, Processes, and Conceptual Model

Seepage is the movement of liquid water into emplacement drifts. The basic conceptual model for seepage is that openings in unsaturated media act as capillary barriers and divert water around them (Figure 3.2-12). This capillary barrier effect has been tested in the ESF by niche (a side tunnel off the main tunnel) tests, in which water is injected above a niche. Results from the tests indicate that most of the water does not seep into an opening less than 1 m (3 ft) below the water-release point (CRWMS M&O 2000 [145774], Sections 2.2.2.2.1, 3.9.4). In addition, no naturally occurring and continuous seeps have been observed so far in the ESF, although the lack of seeps may be partially caused by drying from tunnel ventilation. (A damp feature, with no dripping, was observed immediately after dry excavation of Niche 3566; CRWMS M&O 2000 [145774], Section 2.2.2.2.1).

For seepage to occur in the conceptual model, the fractures at the drift wall must be locally saturated. Drift walls can become locally saturated either by disturbance to the flow field caused by the drift opening or by heterogeneity in the permeability field that creates channeled flow and local ponding. Of the two causes, the heterogeneity effect is more important. Drift-scale flow calculations made with uniform hydrologic properties suggest that seepage will not occur at expected percolation fluxes. However, calculations that include heterogeneity do predict seepage, with the amount depending on the hydrologic properties and the incoming percolation flux (Birkholzer et al. 1999 [105170], Sections 3 and 4). In the seepage conceptual model, there is a percolation flux, called the seepage threshold, below which there is no seepage (that is, all flow is diverted around the drift opening). Another way to state the effect of heterogeneity is that the seepage threshold is lower when the hydrologic properties are heterogeneous than when they are uniform.

Other important factors affecting drift seepage include flow channeling on larger scales (possible focusing of flow above drifts), fracture permeability and capillarity, drift geometry and its degradation over time, the excavation-disturbed zone around the drifts, and the thermal perturbation caused by emplacement of waste (CRWMS M&O 2000 [145774], Sections 3.9.1 and 3.9.3). Capillarity refers to the tendency of a liquid to be held in a pore space or fracture by virtue of its surface tension and adhesion to the solid surface. Capillarity is represented by a model parameter called α (the fracture air-entry parameter), which is proportional to the effective fracture aperture. Capillary forces are weaker when the α parameter is larger, which means that water can seep into a drift more easily when α is larger. The excavation-disturbed zone refers to the region around an excavation where fracture properties are changed from their pre-excavation values. Generally, fractures open up somewhat in the excavation-disturbed zone, and the rock may be broken to create new fractures as well. One result is that fracture permeability is increased significantly in this zone.

As with the modeling of mountain-scale flow, the dual-permeability model would be preferred for modeling seepage at the drift scale. However, to simplify the drift-scale calculations, effects of the rock matrix were not considered, and flow was calculated using a model for only the fracture continuum. This simplification is conservative compared to the dual-permeability model, because the effect of including the matrix would be to decrease the fracture flow and reduce the amount of seepage. (The large capillary suction in matrix pores prevents matrix flow from seeping into drift openings.)

Short-term episodic infiltration transients can be neglected because of the buffering effect of the Paintbrush nonwelded hydrogeologic unit above the potential repository (CRWMS M&O 2000 [145774], Sections 1.2.3, 3.3.6, 3.3.7, 3.7.3.1). However, near-drift transients caused by potential-repository heating, and long-term transients due to climate change, are included in the seepage abstraction for TSPA by taking percolation flux above the drifts from the thermal hydrology model as the input, rather than simply using the percolation flux from the mountain-scale flow model. No credit is taken for seepage reduction due to evaporation and imbibition of water percolating through the hot, dry near-field rock during the thermal pulse. There is additional discussion of thermal effects on seepage in Section 3.3.3. The seepage abstraction also includes effects of drift degradation and flow channeling.

3.2.4.2 Implementation in the Total System Performance Assessment

Modeling of seepage into drifts was done in three steps (see Figure 3.2-13):

1. Conceptual models were tested against data from the niche seepage tests, and calibrated properties were derived for one of the niches (CRWMS M&O 2000 [145774], Section 3.9.4). Fracture permeability is measured by means of air-injection tests, but the capillary parameter α cannot be measured directly. Calibrating the three-dimensional, heterogeneous seepage process model against seepage tests produced a value for the fracture α parameter that implicitly includes discrete-fracture effects and effects of the excavation-disturbed zone around the niche. (Note that the niche used for calibration was in the Topopah Spring middle nonlithophysal hydrogeologic unit, whereas most of the potential repository is to be in the lower lithophysal unit. Also, the niche was excavated with a road header, whereas the emplacement drifts are to be excavated by tunnel boring machine. The effect of these differences on the results, if any, has not been evaluated as yet.)

2. Because of uncertainty and spatial variability in percolation flux and fracture hydrologic parameters, the seepage process model was run for a wide range of parameter inputs (CRWMS M&O 2000 [145774], Section 3.9.5). Also, several other effects were evaluated, including changes in drift shape due to rockfall and possible correlation between fracture permeability and α parameter.

3. A probabilistic seepage abstraction model was developed using the results of the simulations from step 2, including adjustments to account for drift degradation, possible preferential paths provided by degraded rock bolts, possible correlation between fracture permeability and α parameter, and channeling of flow above the drifts (CRWMS M&O 2000 [145774], Section 3.9.6).

The purpose of the seepage abstraction model is to provide estimates of the amount of seepage that might interact with a waste package. Thus, the modeling in steps 2 and 3 focused on a drift segment of length equal to the average waste-package length plus the spacing between waste packages, making the segment length 5.1 m + 0.1 m = 5.2 m (16.7 ft + 0.3 ft = 17 ft). The drifts are 5.5 m (18 ft) in diameter. (See Section 1.7 for additional description of the potential repository design.) In the simulations, the amount of seepage reported included all water that entered the drift, regardless of whether it seeped in at the top, bottom, or side. The seep flow rate is defined to be the rate of water flow into a drift segment. In TSPA simulations, the seep flow rate is treated as a sampled random variable that is both spatially variable and uncertain.

The connections between seepage into drifts and other TSPA model components are shown in Figure 3.2-14. The uncertainty distributions described in Section 3.2.4.3 are used directly by the TSPA model to generate seepage histories. Seepage information is passed by the TSPA model to the other model components as required.

3.2.4.3 Treatment of Uncertainty and Variability

Within the TSPA model, seepage is parameterized by four quantities: the seepage fraction (fraction of waste-package locations that have seepage), the mean seep flow rate for locations with seepage, the standard deviation of seep flow rate for locations with seepage, and a flow-focusing factor that represents effects of flow channeling (because only a subset of fractures are actively flowing). Uncertainty distributions have been developed for these four quantities, and values are sampled from the uncertainty distributions for each realization (Table 3.2-3).

Table 3.2-3. Uncertainty Distributions Used to Represent Seepage

Parameter	Distribution Type
Seepage fraction (fraction of locations with seepage), f_s	Triangular. Minimum, most likely, and maximum are functions of percolation flux (see Figure 3.2-15 in Section 3.2.4.4)
Mean seep flow rate for locations with seepage, μ_{Qs}	Triangular. Minimum, most likely, and maximum are functions of percolation flux (see Figure 3.2-15 in Section 3.2.4.4)
Standard deviation of seep flow rate for locations with seepage, σ_{Qs}	Triangular. Minimum, most likely, and maximum are functions of percolation flux
Flow-focusing factor, F	Log-uniform: 1 to 47 for low-infiltration case 1 to 22 for medium-infiltration case 1 to 9.7 for high-infiltration case

Source: CRWMS M&O 2000 [145774], Sections 3.9.6.3 and 3.9.6.4

In step 3 above, it was concluded that computed seepage primarily depends on the input percolation flux and on the ratio of the geometric mean of the heterogeneous permeability field to the fracture α parameter. Using available data, distributions to represent spatial variability and uncertainty were developed for that ratio. These distributions were combined with the seepage-model results to obtain spatial-variability and uncertainty distributions for seepage as a function of percolation flux (CRWMS M&O 2000 [145774], Section 3.9.6.4). Distributions of the degree of flow focusing above the drifts were determined based on estimates of spacing of actively flowing fractures in the mountain-scale flow model (CRWMS M&O 2000 [145774], Section 3.9.6.3). As indicated in Table 3.2-3, the seepage fraction and seep flow rate distributions are functions of percolation flux; the flow-focusing factor is a constant for a given TSPA realization, but varies for the different infiltration cases.

Seepage is highly variable in space because of both variability in percolation flux and heterogeneity in fracture hydrologic properties. Within each TSPA realization, seep flow rate is distributed according to the quantities in Table 3.2-3. To explain, first consider a simpler situation in which percolation dependence and the flow-focusing factor are neglected. In this situation, the spatial distribution of seep flow rate, Q_s, is described by the seepage fraction f_s, mean seep flow rate μ_{Qs}, and standard deviation of seep flow rate σ_{Qs}. The seepage fraction, f_s, gives the fraction of locations that have nonzero seepage. Locations with nonzero seepage have a local seepage threshold that is lower than the local percolation flux. For the locations with seepage, Q_s is sampled from a distribution with mean μ_{Qs} and standard deviation σ_{Qs}. A beta probability distribution is used because a lower bound of zero can be specified (a normal distribution would not be appropriate because it would include negative flow rates). A beta distribution also requires an upper bound, so an arbitrary large upper bound is specified. The upper bound does not significantly affect the sampling as long as it is well above the sampled values.

The extension to the situation with f_s, μ_{Qs}, and σ_{Qs} as functions of percolation flux is straightforward, requiring only that the values appropriate to the local percolation flux be used at each spatial location. The flow-focusing factor F is included by multiplying the percolation flux at a location by F and then determining f_s, μ_{Qs}, and σ_{Qs}, based on that higher flux. The results must then be adjusted because a higher, focused, percolation flux in one location implies lower flux somewhere else in order to conserve water mass. Flow focusing generally implies lower seepage fractions and higher seep flow rates, with the total amount of seepage higher than it would have been without focusing (CRWMS M&O 2000 [142004], Section 6.3.3.2.1). Details of the sampling method can be found in *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384], Section 6.3.1.3).

As mentioned in Section 3.2.4.1, the percolation flux for input to the seepage abstraction model is taken from the thermal hydrology model in order to approximate thermal perturbations to seepage (as noted in that section, seepage during the thermal period is over predicted because no credit is taken for seepage reduction due to evaporation and imbibition of water percolating through the hot, dry near-field rock). For the TSPA simulations, thermally perturbed percolation histories are available at 610 repository locations (Section 3.3.3). However, engineered-system calculations within the TSPA model are based on 30 environment groups (Section 3.3.2). Quantities such as temperature and relative humidity are averaged over those groups for use by most of the engineered-system models. (The exceptions are waste-package and drip-shield degradation, which use information from the 610 locations directly.) Because of the nonlinear relationship between seepage and percolation flux, seepage histories are generated for all the locations in each environment group and then averaged for use by the other models, rather than averaging the percolation flux and calculating seepage from the averaged percolation history.

3.2.4.4 Results and Interpretation

Figure 3.2-15 shows some results of the seepage modeling. The top graph illustrates the distribution of seepage fraction, or fraction of waste-package locations with seepage (f_s). The bottom graph illustrates the distribution of mean seep flow rate for locations with seepage, which is the average rate of water flow into the drift at locations with seeps (μ_{Qs}). Each graph shows the three quantities that define the triangular uncertainty distribution. At a given percolation flux, there is a minimum value, a most likely value, and a maximum value. The minimum and maximum values define the base of the triangle, and the most likely value defines the location of the peak of the triangle. The graph for standard deviation of seep flow rate (σ_{Qs}) is not shown because it is quite similar to the graph for mean seep flow rate. The standard deviations of seep flow rate are nearly as large as the mean in most cases and even larger than the mean in a few cases. These large standard deviations result from the large differences in seepage from one set of process-model simulations to another and indicate a large spatial variability of seepage. Values of σ_{Qs} (and the other quantities) are tabulated in *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774], Table 3.9-2).

3.2.5 An Alternative Long-Term Climate Definition

The base-case climate model has no climate changes after 2000 years (Section 3.2.1). For the long time period necessary for determination of the peak dose, it is more appropriate to use a climate model that includes full-glacial climates and interglacial climates in addition to the average climate. For this reason, an extended climate model for times up to 1 million years was developed (CRWMS M&O 2000 [153038]). This more detailed climate is used only in Section 4.1.3 of this report. It is based on paleoclimate data in the same way as the 10,000-year climate model is (see Section 3.2.1). The extended climate is summarized in Tables 3.2-4 and 3.2-5, which show the climate sequence and climate analogs, respectively. Some explanation and discussion follows.

Table 3.2-4. Climate Sequence from 10,000 to 1 Million Years

Climate State	Begin Time (years)	End Time (years)
Intermediate/Monsoon	10,000	38,000
Glacial, Stage 8/10	38,000	49,000
Intermediate	49,000	65,000
Interglacial	65,000	77,000
Intermediate/Monsoon	77,000	106,000
Glacial, Stage 8/10	106,000	120,000
Intermediate	120,000	137,000

Table 3.2-4. Climate Sequence from 10,000 to 1 Million Years (Continued)

Climate State	Begin Time (years)	End Time (years)
Interglacial	137,000	148,000
Intermediate/Monsoon	148,000	200,000
Glacial, Stage 6/16	200,000	213,000
Intermediate	213,000	229,000
Interglacial	229,000	241,000
Intermediate/Monsoon	241,000	291,000
Glacial, Stage 4	291,000	329,000
Intermediate	329,000	345,000
Interglacial	345,000	355,000
Intermediate/Monsoon	355,000	401,000
Glacial, Stage 8/10	401,000	409,000
Intermediate	409,000	422,000
Interglacial	422,000	432,000
Intermediate/Monsoon	432,000	471,000
Glacial, Stage 8/10	471,000	482,000
Intermediate	482,000	497,000
Interglacial	497,000	507,000
Intermediate/Monsoon	507,000	555,000
Glacial, Stage 6/16	555,000	595,000
Intermediate	595,000	611,000
Interglacial	611,000	622,000
Intermediate/Monsoon	622,000	672,000
Glacial, Stage 4	672,000	688,000
Intermediate	688,000	704,000
Interglacial	704,000	715,000
Intermediate/Monsoon	715,000	761,000
Glacial, Stage 8/10	761,000	788,000
Intermediate	788,000	801,000
Interglacial	801,000	811,000
Intermediate/Monsoon	811,000	854,000
Glacial, Stage 8/10	854,000	864,000
Intermediate	864,000	877,000
Interglacial	877,000	887,000
Intermediate/Monsoon	887,000	934,000
Glacial, Stage 6/16	934,000	957,000
Intermediate	957,000	970,000
Interglacial	970,000	981,000
Intermediate/Monsoon	981,000	1,000,000

Source: CRWMS M&O 2000 [153038], Table 6-6

Table 3.2-5. Analog Sites for Extended-Climate States

Climate State	Representative Meteorological Stations	Substitutions
Interglacial Climate (same as Present-Day)	Yucca Mountain and vicinity	
Intermediate Climate (same as Glacial-Transition)	Upper Bound: Spokane, Washington Rosalia, Washington St. John, Washington	
	Lower Bound: Beowawe, Nevada Delta, Utah	
Intermediate/Monsoon Climate	Oscillations between Intermediate and Monsoon climate states	Intermediate Climate is used as a substitute for Intermediate/Monsoon
Glacial Climate, Stage 8/10	Upper Bound: Chewelah, Washington	Lake Yellowstone, Wyoming is used as a substitute for Chewelah, Washington
	Lower Bound: Spokane, Washington Rosalia, Washington St. John, Washington	
Glacial Climate, Stage 6/16	Upper Bound: Lake Yellowstone, Wyoming	
	Lower Bound: Browning, Montana Simpson, Montana	
Glacial Climate, Stage 4	Upper Bound: Browning, Montana Simpson, Montana	Lake Yellowstone, Wyoming is combined with Browning and Simpson, Montana for this case
	Lower Bound: Elko, Nevada	Simpson, Montana is used as a substitute for Elko, Nevada

Source: Analogs in column 2 from CRWMS M&O 2000 [153038], Table 6-3 ; substitutions in column 3 are approximations made for this analysis

Six climate states are specified for the post-10,000-year climate sequence: interglacial, monsoon, intermediate, and three different full-glacial climates. The present-day climate is an interglacial climate, and present-day climate is used as an analog for future interglacial climates as well. The intermediate climate is the same as the glacial-transition climate (CRWMS M&O 2000 [153038], Section 6.1). Preceding each glacial state in the climate sequence is a period called intermediate/monsoon, when the climate is alternating between those two states (CRWMS M&O 2000 [153038], Table 6-6). For TSPA simulations, the climate sequence was simplified by lumping together the alternating intermediate and monsoon climates into a single intermediate climate state. This approximation is acceptable for TSPA because monsoon climate is estimated to occur for less than 7 percent of the intermediate/monsoon climate state (CRWMS M&O 2000 [153038], Section 7).

Past climates are described with a terminology that refers to different climate periods as stages (which are based on oxygen isotope ratios from marine carbonate records). The current interglacial period is stage 1, the most recent glacial period was stage 2, the interglacial period before that was stage 3, and so on. In developing the post-10,000-year climate sequence, stages 1–3 were combined because that provides a better fit to the Devils Hole and Owens Lake data (CRWMS M&O 2000 [153038], Section 6.4.5). The future-climate sequence was developed by observing that the stage 11 interglacial period was similar to the stage 1 interglacial period or the combination stages 1–3 period. Thus, the expectation is that the next glacial period in our future will be like stage 10, the glacial period after that like stage 8, etc. (CRWMS M&O 2000 [153038], Section 6). Glacial stages 8 and 10 were similar, so the same analogs are specified for them in the climate sequence; similarly, stages 6 and 16 were similar (CRWMS M&O 2000 [153038], Section 6.3).

As indicated in Table 3.2-5, three substitutions were made in the full-glacial analogs for TSPA simulations: Lake Yellowstone, Wyoming, was used to represent the stage-8/10 upper-bound climate instead of Chewelah, Washington; Simpson, Montana, was used to represent the stage-4 lower-bound climate instead of Elko, Nevada; and Lake Yellowstone, Wyoming was combined with Browning and Simpson, Montana for the stage-4 upper-bound climate. The first two substitutions were made because infiltration results were not available for the Chewelah and Elko climate analogs. The substitutions are conservative in that the substituted sites all have greater effective moisture (some combination of lower average annual temperature and greater average annual precipitation) than the originally specified sites: Simpson has mean annual precipitation higher than Elko and mean annual temperature lower than Elko (CRWMS M&O 2000 [153038], Table 6-3), both of which tend to increase net infiltration. Lake Yellowstone has mean annual precipitation slightly lower than Chewelah, but has significantly lower mean annual temperature, which should result in greater net infiltration for the Lake Yellowstone analog. The third substitution was made so that the stage-4 glacial period would be colder and wetter than the intermediate climate in the TSPA model. The third substitution is also clearly conservative because Lake Yellowstone is much wetter than Browning and Simpson.

Net infiltration maps were generated for the additional climate analogs using the same method as described in Section 3.2.2. The results are summarized in Table 3.2-6, which shows the repository-average net infiltration for the glacial-climate cases.

Table 3.2-6. Average Net Infiltrations for the Glacial Infiltration Cases

Infiltration Case	Glacial Climate, Stage 8/10	Glacial Climate, Stage 6/16	Glacial Climate, Stage 4
Low infiltration	37 mm/yr	28 mm/yr	17 mm/yr
Medium infiltration	71 mm/yr[a]	71 mm/yr[a]	28 mm/yr[b]
High infiltration	110 mm/yr	110 mm/yr	71 mm/yr

Source: CRWMS M&O 2000 [153002], file OIS16-results2.doc

NOTES: 25.4 mm = 1 in.

[a]Average of high and low cases for glacial climate, stage 6/16

[b]Low case for glacial climate, stage 6/16

One aspect of the method described in Section 3.2.2 was simplified for this analysis. For monsoon and glacial-transition climates, the medium-infiltration case was generated by averaging the low- and high-infiltration cases at each location in the infiltration maps (see Section 3.2.2.3). The same approach was followed for glacial climate, stage 6/16. However, in order to reduce the number of new UZ-flow simulations needed, the medium-infiltration cases for glacial climate stages 8/10 and 4 were replaced by cases already available. For stage 8/10, the medium-infiltration case for stage 6/16 was used, and for stage 4, the low-infiltration case for stage 6/16 was used.

The extended climate model is illustrated in Figure 3.2-16, which shows repository-average net infiltration over time up to 1 million years for the three infiltration cases. The first 10,000 years in this figure are the same as Figure 3.2-7. The part of the figure after 10,000 years combines the climate sequence in Table 3.2-4 with the average infiltrations in Tables 3.2-2 and 3.2-6.

Mountain-scale UZ flow fields were generated for four new cases for this extended climate analysis: glacial-climate, stage-6/16, low-infiltration case, glacial-climate, stage-6/16, medium-infiltration case, glacial-climate, stage-6/16, high-infiltration case, and glacial-climate, stage 4, low-infiltration case. All other cases are represented either by these or by a glacial-transition case that was already available. For calculation of UZ transport within the TSPA simulation, the UZ flow fields are used in sequence, with the sequence defined by Tables 3.2-1 and 3.2-4.

New thermal hydrology results were not generated for the extended climate analysis. The extended climate sequence does not diverge from the base-case climate sequence until 38,000 years, by which time temperatures are near ambient, so changes in thermal hydrology from the base case would be very small. However, the additional climate changes are important to seepage and, as mentioned in Section 3.2.4.2, in the TSPA nominal base case, percolation flux above the drifts is taken from the thermal hydrology model for input to the seepage abstraction within the TSPA model. Since the climate changes after 38,000 years are not included in the thermal hydrology, it is necessary to use a different percolation input to the seepage abstraction at later times. This is done by treating the spatial distribution of infiltration for each climate state and infiltration case as a probability distribution, from which values are sampled for input to the seepage abstraction. The sampling is correlated with the percolation values for the glacial-transition climate so that a given location remains either wet or dry over time. As with the base case, this procedure is done separately for the five infiltration bins described in Section 3.3.2. Note that the infiltration bins are still based on glacial-transition climate since it is in effect most of the time (intermediate climate is the same as glacial-transition).

3.3 ENGINEERED BARRIER SYSTEM ENVIRONMENTS

EBS environments refer to the thermodynamic and chemical environments within the emplacement drifts. These environments form the background on which the engineered components of the potential repository evolve. The environmental characteristics discussed in this section include degradation of the drift itself (including rockfall into the drifts), temperature, RH, liquid saturation, pH, and liquid and gas composition at various locations within the drifts. Thermal effects on flow and chemistry outside the drifts are also discussed, as they affect the amount and composition of water and gas entering the drifts.

The EBS environments are important to the potential repository performance to the extent that they help determine degradation rates of the engineered barrier components, quantities and species of mobilized radionuclides, and transport of radionuclides and fluids through the drift into the UZ. The principal performance-related engineered barrier components that initially prevent water contact with waste forms and determine fluid transport paths are the drip shield and the waste package. Waste form degradation determines quantities and species of mobilized radionuclides. The EBS chemical environments also affect radionuclide solubility, colloid loading with radionuclides, and colloid stability, all of which affect the mobile radionuclide source term for transport. Fluid transport paths and rates, coupled with the source term, determine radionuclide release rates to the UZ. Environmental effects on waste-package and drip-shield degradation are discussed in Section 3.4; environmental effects on waste-form degradation and radionuclide mobilization are discussed in Section 3.5; and environmental effects on fluid and radionuclide transport are discussed in Section 3.6.

The following sections discuss drift degradation, grouping of waste packages within the TSPA model, thermal hydrology, and in-drift chemistry. These aspects of the EBS environments are discussed in terms of submodels that serve as bases for quantitative evaluations and predictions. The relationships of the EBS environmental submodels with each other and with other TSPA components are diagrammed in Figure 3.3-1.

3.3.1 Drift Degradation

This section describes the deterioration of the rock mass surrounding the potential repository emplacement drifts. Deterioration occurs by failure of fractures that bound blocks of rock at the drift walls and falling of those blocks into the drift. The description of deterioration is given in terms of a key block analysis (CRWMS M&O 2000 [151635]), which is a tool used for the following purposes:

- Provide a statistical description of block sizes formed by fractures around the emplacement drifts
- Estimate changes in drift profiles due to fallen blocks of rock
- Provide an estimate of the time required for significant drift deterioration to occur.

Drift degradation has the potential to affect drip shield integrity and thermal-hydrologic environments in drifts. The results of the drift degradation model and analysis were used to assess those effects, and they were found to have negligible impact for TSPA calculations. Results and an interpretation are given in Section 3.3.1.4.

3.3.1.1 Features, Processes, and Conceptual Model Related to Drift Degradation

Key blocks are formed at the surrounding rock mass of an excavation by the intersection of three or more planes of structural discontinuities, as shown in Figure 3.3-2, (taken from CRWMS M&O 2000 [151635], Figure 1). Key blocks can become dislodged (termed block failure) and fall because of seismic thermal effects and time-dependent effects. Those features and processes constitute the conceptual model for drift degradation. Quantification of drift degradation by block failures and falls is the output of a key block analysis (CRWMS M&O 2000 [151635]). Abstracted results from that analysis are given in Section 3.3.1.4.

The conceptual model and analysis are supported by the following field observations of these features, which are documented in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 6.2). Sources of mapping information, including observation stations, are also described in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 4.1). Key blocks in the 5-m diameter cross drift are first evident in the crown at about Station 10+50 in the Topopah Spring Tuff crystal-poor middle nonlithophysal zone (Tptpmn) unit. Most of the key blocks in this region are of minor size and, typically, fall immediately after excavation, prior to ground support installation. Key blocks are possible in this area because of the increased presence of the plane of weakness (i.e., a vapor-phase parting) in the near-horizontal orientation that intersects with two opposing near-vertical joint planes. Fallout from these key blocks during excavation is typical of the rock in the middle nonlithophysal zone (Tptpmn) of the TSw2 thermal-mechanical unit. The largest resultant void is about 0.5 m^3 at approximately Station 11+55. No unstable key blocks were observed in the field. For a modeled drift azimuth of 45 degrees, which corresponds most closely with the Cross Drift azimuth of 49 degrees (CRWMS M&O 1998 [113521], p. 10), 75 percent of the blocks simulated were less than 0.27 cubic meters, and 90 percent of the blocks simulated were less than 1.15 cubic meters (CRWMS M&O 2000 [151635], Table 12). The maximum block observed in the Cross Drift is captured within this range. Modeled blocks occur most frequently in the Tptmn unit (CRWMS M&O 2000 [151635], Table 15), which agrees with field observations.

The approach to this probabilistic key block drift degradation analysis, provided in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 6.3), involves the following:

- Analyze blocks, and their associated joints, that have fallen in the field
- Collect and assess joint geometrical data and joint frictional properties data from the ESF main loop and Enhanced Characterization of the Repository Block Cross Drift (CRWMS M&O 2000 [151635], Section 3.2) to develop the joint modeling inputs for DRKBA software (CRWMS M&O 2000 [149991])
- Analyze the joint data to assess the potential formation of key blocks using DRKBA, including the maximum block size
- Analyze the seismic and thermal effects on joint and block movement
- Analyze the DRKBA block-size distribution data for each of the four lithologic unit within the potential repository host horizon
- Determine the number and average volume of rockfall per unit length of drift for various levels of seismic hazard

- Evaluate postclosure frequency of block failure for 10,000 years
- Analyze the drift profile, showing the progressive movement of joints and blocks, with time.

3.3.1.2 Integration of Drift Degradation within Total System Performance Assessment

Effects of drift degradation (rockfall) on drip shield breaching and thermal hydrology effects were identified as having potential impact on TSPA calculations and were evaluated. Evaluation results are given in Section 3.3.1.4.

3.3.1.3 Treatment of Uncertainty

The representativeness of the DRKBA rock fall model is affected by how well the fracture data inputs describe the actual fracture conditions at the emplacement drift horizon. The natural variability of fractures within a rock mass always represents uncertainty in the design of structures in rock. The uncertainty of fracture geometry (i.e., the orientation, spacing, and trace length of fractures) anticipated at the emplacement drift horizon has been diminished with the construction and subsequent detailed fracture mapping of the approximately 10 km of tunnels that comprise the ESF. Two approaches were used to collect fracture data in the ESF: full-periphery geologic mapping and detailed line surveys. The full-periphery geologic mapping presents a full-periphery map of the tunnel walls as encountered in the subsurface excavation. The detailed line survey presents a detailed inventory of fractures and associated attributes that intersect the survey line along the tunnel wall. The vast amount of fracture data collected from the north-south-trending Main Drift and the east-west-trending Cross Drift provide a very good representation of the range of fractures anticipated at the emplacement drift horizon.

The degree of fracture characterization at Yucca Mountain far exceeds that done in typical tunneling projects throughout the world. Typically, tunneling projects will rely on a series of boreholes to characterize fractures anticipated at the tunnel horizon. Because of the limited exposure of the rock mass when using borehole information, the uncertainty associated with predictions from fracture geometry data for typical tunneling projects can be considerably greater than that in the DRKBA rock fall model for the Yucca Mountain Project. The fracture mapping of the tunnels in the ESF provides a high degree of confidence that the full range of fracture characteristics anticipated at the emplacement drift horizon has been sampled. Because the full-periphery geologic mappings depict an unrolled view of the full periphery of the subsurface excavation, they represent a true volume sampling of the rock mass. This range of fracture variability from tunnel mapping has been captured in the DRKBA rock fall model through multiple Monte Carlo simulations of the rock mass, thus providing a realistic representation of the fractured rock mass at the emplacement drift horizon.

To assess the uncertainties associated with the seismic component of the DRKBA rock fall model, alternative analytical methods involving the dynamic functions of the distinct element code UDEC (CRWMS M&O 2000 [151635], Section 6.3.1) were used to verify the approach. Furthermore, the seismic component of the DRKBA rock fall model is validated by comparing the results of the model (CRWMS M&O 2000 [151635]) to case histories of various underground structures subjected to earthquakes. The seismic effect on both the size and number of blocks in the DRKBA rock fall model is relatively minor, which is consistent with the case histories.

Another source of uncertainty in the DRKBA rock fall model is the effect of thermal loading on block development. Due to the lateral confinement of the rock, the predicted thermal stress is highest in the horizontal direction. The high horizontal thermal stress will likely provide a locking effect for the blocks formed by the predominant vertical joint sets during the heating period, thus potentially reducing the size and number of blocks developed. However, this locking effect has been conservatively ignored in the DRKBA rock fall model. The DRKBA approach assumes that time-dependent and thermal degradation occurs mainly due to the reduction of joint cohesion. Joint cohesion exists due to the asperities along the joint surface. These asperities may shear off with time and they may shear off due to the increase shear stress caused by the thermal effect. To account for uncertainties associated with the thermal and time-dependent effects on rock fall, a conservative reduction of the joint cohesive strength has been included in the model. Shear strength experiments using core from the emplacement drift horizon resulted in a cohesion value of 0.86 MPa with a standard error of ±0.81 Mpa (CRWMS M&O 2000 [151635], Section 7). Due to this wide error range in laboratory values, the joint cohesion in the DRKBA rock fall model was conservatively initialized as 0.1 MPa. Over a period of 10,000 years, joint cohesion was further reduced to 0.01 MPa.

3.3.1.4 Abstracted Results and Interpretation

Predictions of key blocks forming at the performance confirmation drifts and emplacement drifts located in the four lithologic units were calculated. The results are presented for both a static key block assessment and a quasi-static key block assessment to account for seismic, thermal, and time effects on key blocks. The method used for the quasi-static analysis to simulate the seismic effect is described in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 6.3.4). Three levels of earthquake, representing a 1,000-year event (Level 1), a 5,000-year event (Level 2), and a 10,000-year event (Level 3), are considered. The License Application Design Selection drift orientation, with an azimuth of 105°, is the designated orientation for the quasi-static analysis.

In the DRKBA analysis, random joint patterns are generated, with joint centers positioned in three-dimensional space, considering each joint set in sequence for each Monte Carlo simulation. The forming of key blocks is, therefore, different in each Monte Carlo simulation. Test runs were conducted to determine an adequate number of Monte Carlo simulations for the analyses, as described in Attachment IV of the reference AMR. Based on the test-run results, 200 Monte Carlo simulations are adequate for the Tptpmn unit and 400 Monte Carlo simulations are adequate for the rest of the units.

Predicted numbers and average volumes of key blocks per unit length (km) along emplacement drift with seismic consideration are given in Tables 3.3-1 and Table 3.3-2, respectively (taken from CRWMS M&O 2000 [151635], Tables 20 and 21). Small seismic-effects on numbers and volumes of key blocks are shown.

Table 3.3-1. Predicted Number of Key Blocks per Unit Length along Emplacement Drift, with Seismic Consideration

Lithologic Unit	Static (km^{-1})	Static Plus Seismic (km^{-1})		
		Level 1	Level 2	Level 3
Tptpul	15	15	17	17
Tptpmn	37	38	40	40
Tptpll	3	3	3	3
Tptpln	3	3	5	5

Source: CRWMS M&O 2000 [151635], Table 20

Table 3.3-2. Predicted Average Volume of Key Blocks per Unit Length along Emplacement Drift, with Seismic Consideration

Lithologic Unit	Static (m^3/km)	Static Plus Seismic (m^3/km)		
		Level 1	Level 2	Level 3
Tptpul	4.9	5.0	9.5	12.1
Tptpmn	18	25.9	32.3	32.3
Tptpll	0.8	0.8	0.8	0.8
Tptpln	3.0	5.5	14.4	14.4

Source: CRWMS M&O 2000 [151635], Table 21

Block volumes corresponding to various levels of predicted cumulative frequency of occurrence are also given in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 6.4.1.1).

An analysis was done using an approach that accounts for the time-dependent and thermal effect on joint cohesion degradation. The development and justification of this approach is described in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 6.3.5 and Attachment VI). Four different times are selected for the analysis: 0 years (static condition), 200 years, 2,000 years, and 10,000 years. The reduction of joint cohesion was calculated, ranging from an initial value of 0.1 Mpa to 0.02 Mpa at year 200 and 0.01 Mpa at years 2,000 and 10,000. Note that the reduction of joint cohesion is predicted to be very small in the period between years 2,000 and 10,000.

The predicted number of key blocks per kilometer of drift for the License Application Design Selection orientation is listed in Table 3.3-3. Only minor increases of key blocks are predicted between year 200 and year 2,000. No change is predicted from year 2,000 to year 10,000. The predicted average volume of rockfall per unit length of drift is listed in Table 3.3-4.

Table 3.3-3. Predicted Number of Key Blocks per Unit Length along Emplacement Drift, with Time-Dependent and Thermal Consideration

Lithologic Unit	Static (km^{-1})	Year 200 (km^{-1})	Year 2,000 (km^{-1})	Year 10,000 (km^{-1})
Tptpul	15	14	15	15
Tptpmn	37	37	39	39
Tptpll	3	4	4	4
Tptpln	3	4	5	5

Source: CRWMS M&O 2000 [151635], Figure 23

Table 3.3-4. Predicted Average Volume of Fallen Key Blocks (predicted average volume of rock fall) per Unit Length along Emplacement Drift, with Time-Dependent and Thermal Consideration

Lithologic Unit	Static (m^3/km)	Year 200 (m^3/km)	Year 2,000 (m^3/km)	Year 10,000 (m^3/km)
Tptpul	4.9	5.1	8.8	8.8
Tptpmn	18	15.8	19.0	19.0
Tptpll	0.8	0.8	0.9	0.9
Tptpln	3.0	3.4	8.4	8.4

Source: CRWMS M&O 2000 [151635], Figure 24

The results indicate that time-dependent and thermal effects have a minor impact on rockfall.

Degraded, worst-case drift cross-section profiles were also predicted, and results are given in *Drift Degradation Analysis* (CRWMS M&O 2000 [151635], Section 6.4.3). Most of the emplacement drift openings were not affected by rockfall. The highest percentage of drift affected by rockfall was 16 percent in the Tptpmn unit. The Tptpmn unit also produced the largest volume of key blocks per kilometer.

More detailed block volume results are provided in the Supporting Rock Fall Calculation for Drift Degradation: Drift Reorientation with No Backfill (CRWMS M&O 2000 [149639], Section 6.4).

Effects of the above drift degradation (rockfall) results on drip shield breaching and thermal hydrologic effects were assessed and found to be negligible for TSPA calculations. The potential for cracking of a drip shield because of the impact of a rock on it was assessed and found to be negligible (CRWMS M&O 2000 [147396]). Rocks as large as 52 metric tons were considered.

The transient, dynamic response of the DS to impact from a large rock block has been evaluated with a detailed finite-element model (CRWMS M&O 2000 [149574]). These analyses considered blocks weighing 2 MT, 4 MT, 6 MT, 8 MT, and 52 MT and also considered on-center and off-center rock falls. The drip shield does not fail catastrophically for either the on-center or the off-center impacts (CRWMS M&O 2000 [149574], Figures III-7 and III-15). In fact, it is seen to retain its basic "mail box" shape. However, there is some permanent deformation. For example, the permanent displacement of the crown of the drip shield is approximately 70 mm for a 6 MT block and 80 mm for a 52 MT block. (CRWMS M&O 2000 [149574], Figures III-10 and III-16).

The finite-element analyses are conservative, particularly near the ends of the drip shield, because the overlap between adjacent drip shields are ignored. The overlap will tend to reduce the permanent displacement of the crown locally, providing added stability for a part of the drip shield that is critical as a flow barrier. The degree of conservatism in ignoring the overlap has not been quantified.

A second effect from rock fall has also been considered in these analyses. The permanent deformation from the rock fall generates stress concentrations and damaged regions with a potential for SCC. The number of potential cracks per drip shield has been predicted from the detailed stress fields in the finite-element calculations (CRWMS M&O 2000 [149574], Table 6-1). In theory, these cracks can form advective pathways for liquid flow that can bypass the drip shield. In practice, a dense passive corrosion oxide film is expected to fill any through-wall SCCs in Titanium Grade 7 that form after a rock fall (CRWMS M&O 2000 [151949], Item No. 4). In addition, evaporation of J-13 water slowly flowing through such a crack will lead to rapid scaling by calcite deposition. The ingrowth of corrosion products and the calcite deposition are expect to result in a very tightly plugged crack that cannot pass a significant advective flux through the drip shield (CRWMS M&O 2000 [151949]).

The drip shield will continue to function properly provided that the deflection of the crown does not cause a continuous fluid pathway to form between adjacent drip shields. The permanent vertical displacements from rock fall are small relative to the overlap between adjacent drip shields, 595 mm (see Attachment I). In addition, the basic shape of the drip shield is essentially unchanged for the impact of even the largest blocks. It is therefore unlikely that a rock fall will damage the drip shield to the point that it will not function as a flow barrier.

The effect of rock fall on the as-emplaced drip shield has therefore been screened out from the TSPA-SR. In the as-emplaced condition, there appears to be an adequate margin of safety to ensure that the deflection of the crown does not cause a continuous fluid pathway to form and that SCCs will not provide significant advective pathways through the drip shield. The effect of rock fall has also been screened out as the drip shield degrades, pending more complete analyses of drip shield response.

The effect of rock fall on thermal hydrology is likely to be negligible, because the calculated volume of fallen rock is small (about 0.1 percent) compared with the unoccupied drift volume of 16,650 cubic meters per kilometer of emplacement drift (drift volume outside of the drip shield and above the invert). The values of in-drift geometry and dimensional data used to calculate the unoccupied drift volume were taken from Figure 2 and Table 2 in *Tabulated In-Drift Geometric and Thermal Properties Used in Drift-Scale Models for TSPA-SR* (CRWMS M&O 2000 [151014]).

3.3.2 Environmental Groups

Within the TSPA model, most engineered-system calculations are performed for a limited number of waste package locations. In the model, each of these locations is representative of a group of waste packages with similar environmental characteristics. Radionuclide releases, for example, are calculated for a representative waste package and then scaled up by the number of failed waste packages in the group. (Note that the waste packages in a group do not all fail at the same time, because additional variability is included in the waste package degradation calculation.)

For the VA, waste package groups were based on physical location (six potential repository subregions), waste type (CSNF, codisposal waste, or DSNF), and seepage (always exposed to seepage, exposed to seepage during the wettest two climates, exposed to seepage only during the wettest climate, or never exposed to seepage) (CRWMS M&O 1998 [100371], Section 11.2.1.3). For the SR, the waste package groups are based on infiltration rather than physical location, because radionuclide dissolution and release depend more directly on infiltration than on the location within the potential repository. The other two discriminators are similar to before, though with fewer subdivisions.

Infiltration Bins–Potential repository locations are grouped according to the infiltration at the surface above each location during the glacial-transition climate. Five infiltration bins are defined: 0 to 3 mm/yr (0 to 0.1 in./yr), 3 to 10 mm/yr (0.1 to 0.4 in./yr), 10 to 20 mm/yr (0.4 to 0.8 in./yr), 20 to 60 mm/yr (0.8 to 2.4 in./yr), and 60+ mm/yr (2.4+ in./yr). Glacial-transition infiltration is used because the glacial-transition climate is in effect most of the time during a TSPA simulation, and it is in effect during later times when radionuclide releases are more likely. In principle, it would be preferable to use local percolation flux as the discriminator rather than surface infiltration, but the difference between the two is not large (see Figure 3.2-8). The division of the potential repository into the five bins for each of the three infiltration cases (Section 3.2.2) is shown in Figure 3.3-3. The potential repository footprint in this figure is somewhat simplified and is not quite the same as in the final Site Recommendation design. Waste packages in low-infiltration realizations are all in the first two bins (i.e., between 0 and 10 mm/yr, or 0 and 0.4 in./yr), waste packages in medium-infiltration realizations are mostly in the third and fourth bins (but with some in the other bins as well), and waste packages in high-infiltration realizations are mostly in the fourth and fifth bins (but with some in the second and third bins). Additional details can be found in the *Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204], Section 6.2) and *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384], Section 6.3.1.3). Infiltration (or percolation flux) affects the local temperature, RH, seepage, chemistry, and radionuclide transport.

Waste Type–Waste packages are divided into two types: commercial spent nuclear fuel and codisposal waste. Codisposal waste consists of a combination of vitrified HLW and DSNF in a waste package. In addition to having different radionuclide inventories, the waste type affects the local temperature, RH, seepage, and chemistry, because codisposal waste packages tend to be somewhat cooler than CSNF packages.

Seepage State–Waste package locations are grouped according to whether they have seepage at all times, some of the time, or none of the time. (When there is no seepage at a particular location and time, it can be said that the local percolation flux is below the local seepage threshold.) The seepage state affects the seepage flux, chemistry, and in-drift radionuclide transport. In principle, seepage should affect the temperature and RH as well, but that back-coupling was neglected in the TSPA model because it is significant only at very early times.

Five infiltration bins times two waste types times three seepage states leads to 30 environmental groups that are used to describe conditions for radionuclide mobilization, release, and in-drift transport (Sections 3.5 and 3.6). Spatial variability is incorporated into those models by means of the environmental groups. These 30 groups are adequate to capture the most important aspects of variability within the potential repository—variability in the amount of in-drift water flow and in the type of waste, with the variability in thermal hydrologic and chemical environments in the TSPA model included consistently.

3.3.3 Thermal Hydrologic Environments and Seepage Chemistry

The thermal hydrologic environments consist primarily of the temperature and RH at various locations within the emplacement drifts. In addition, the in-drift chemistry models require information about gas and liquid composition and flow rates, and much of that information comes from the thermal hydrology models.

The following sections discuss the important processes and assumptions, the implementation for use in TSPA simulations, how uncertainty and variability are treated in TSPA simulations, and some results and interpretations. Additional information on Yucca Mountain thermal hydrology and the in-drift environments, including full description and justification of the models and data, can be found in three Process Model Reports and associated Analysis Model Reports: *Near-Field Environment Process Model Report* (CRWMS M&O 2000 [153178]), *Engineered Barrier System Degradation, Flow, and Transport Process Model Report* (CRWMS M&O 2000 [145796]), and *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774]).

3.3.3.1 Features, Processes, and Conceptual Model

The potential Yucca Mountain repository will hold radioactive wastes that emit a large amount of heat from radioactive decay. This heat will cause a significant disturbance in the potential repository environment for thousands of years after waste emplacement. The radioactive decay heat causes temperatures to rise and the subsequent vaporization of *in situ* liquid water, transport of water vapor away from the heat source, condensation of water vapor in cooler regions, and condensate flow driven by gravity and capillary forces. Thermal gradients affect gas and liquid flow in the rock matrix and in fractures, and flow between the fractures and the rock matrix.

After waste emplacement, heat will flow from the potential repository by conduction through the rock and by advection, which means by movement with the gas and water. Through these processes, there will be movement of water vapor, liquid water (both ambient percolating water and heat-mobilized condensate), and heat. Heating will result in elevated temperatures and drying in both the surrounding host rock and regions well away from the potential repository. The progression of thermal hydrologic processes through time is pictured in Figure 3.3-4. Thermally driven mechanical and chemical processes can also potentially cause permanent changes to the fluid-flow characteristics of the rock (e.g., by changing permeability and porosity).

Features, events, and processes (FEPs) regarding thermal hydrology are summarized in *Near Field Environment Process Model Report* (CRWMS M&O 2000 [153178], Section 2.5 and Appendix C) and discussed in detail in *Features, Events, and Processes in Thermal Hydrology and Coupled Processes* (CRWMS M&O 2000 [142895]). Additional information on in-drift FEPs is in *Engineered Barrier System Degradation, Flow, and Transport Process Model Report* (CRWMS M&O 2000 [145796], Sections 1.5.1 and 2.4). Additional information on thermal effects on seepage and far-field flow and transport is in *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774], Section 1.2.3). Many of the FEPs are standard processes that are fully included in the TSPA model, such as "repository dry out due to waste heat" and "resaturation of the repository." Other FEPs are not included in the TSPA model (i.e., they are excluded), typically on the basis of low consequence to the expected dose (e.g., "changes in fluid saturations in the excavation disturbed zone;" [CRWMS M&O 2000 [142895], Section 6.2.15). A complete list of primary FEPs and their status (whether included or excluded) is given in Appendix B.

3.3.3.1.1 Thermal Processes at Multiple Scales

While thermally driven processes are most intense within, and in the immediate vicinity of, the emplacement drifts, larger-scale processes must be considered as well, because they affect the drift environments. Thermal processes at the drift scale (a few meters to tens of meters, or several feet to several tens of feet) include conduction and advection, plus radiant heat transfer when the drifts are open (not backfilled). To these are added all the processes associated with liquid and gas flow. Some of the important thermal hydrologic processes are illustrated in Figure 3.3-5. Additional important considerations at the mountain scale (hundreds of meters to thousands of meters, or several hundred feet to several thousand feet) include the surface topography (thickness of the overburden is a factor in how quickly a location cools off), stratigraphy of the UZ (thermal and hydrologic properties can be quite different across stratigraphic units), spatial variations in infiltration (higher infiltration leads to lower temperature and higher RH), and faster cooling of the potential repository edge as compared with its center.

Consistent with the mountain-scale UZ flow model, the thermal hydrology models use the dual-permeability conceptual flow model, in which flow is modeled as being through two interacting continua, with each continuum being assigned its own spatially variable hydrologic properties, such as permeability and porosity. In general, the fractures are modeled as a highly permeable continuum having low porosity, and the matrix is modeled as a much less permeable continuum having higher porosity. Fracture-matrix interaction is represented with an active-fracture model, in which only a portion of the fractures is actively flowing under unsaturated conditions (CRWMS M&O 2000 [149862], Section 6.3; CRWMS M&O 2000 [145774], Section 3.3.4).

Hydrologic and thermal properties vary among hydrogeologic units, but are uniform within each unit in the thermal hydrology models. As discussed in Section 3.2.4.1, seepage into drifts does not occur in models with uniform hydrologic properties. As a result, seepage does not occur in the thermal hydrology models. If seepage were included, RH would be higher and temperature would be lower at locations where seepage occurs. However, as shown in Section 4.1.2, there is no seepage at almost 90 percent of waste-package locations in the TSPA model. And at the small fraction of locations with seepage, the effect on RH and temperature would be significant only at very early times.

Climate change is included by changing the infiltration boundary condition at prescribed times. The climate is initially the present-day climate, switching to the monsoon climate at 600 years and to the glacial-transition climate at 2,000 years (Section 3.2.1). Climate-induced rise of the water table is not included in the thermal hydrology models, because the water table location is not expected to have a significant impact on the computed in-drift environments.

In addition to the sources of hydrologic data mentioned in Section 3.2, information on thermal properties and processes has come from laboratory tests and from a series of *in situ* thermal tests in the ESF at Yucca Mountain (CRWMS M&O 2000 [153178], Section 3.6).

3.3.3.1.2 Chemical and Mechanical Coupled Processes

Coupled processes refer to the coupled effects of temperature, hydrology, mechanics, and chemistry. When heated, the rock around the emplacement drifts will expand. This thermal expansion will cause fractures to open and close, which, in turn, will change the permeability and porosity of the rock and affect movement of gas and liquid. These changes will last only while the rock is heated. In addition, the thermal stresses can potentially cause rock blocks to move (shear displacements) and new fractures to form, which, again, can affect permeability, porosity, and fluid flow. Unlike the simple thermal-expansion effects, these changes to the rock hydrologic properties can be permanent. All of these effects are called thermal-hydrologic-mechanical coupled processes. Heating also can cause chemical changes in the host rock, such as mineral dissolution in some regions and mineral precipitation elsewhere. Mineral dissolution and precipitation can potentially alter the hydrologic properties of the rock (including fractures) permanently, which would in turn alter fluid flow. These effects are referred to as thermal-hydrologic-chemical (THC) coupled processes. The types of fracture changes that can be caused by coupled processes are illustrated in Figure 3.3-6.

Changes in fracture permeability around an emplacement drift because of thermal-mechanical effects have been calculated (CRWMS M&O 2000 [153178], Section 3.5). The effects are primarily within about two drift diameters (11 m, or 36 ft) of the drifts. Since the drift spacing is much larger than that (81 m), effects on large-scale flow and transport are not expected to be significant. Thermal mechanical effects on drift-scale flow (in particular, on seepage into drifts and radionuclide transport out of the drifts) are neglected in TSPA simulations as well. These effects are thought to be of little importance (CRWMS M&O 2000 [142895], Section 5.2.3). Further evaluation of thermal mechanical effects is ongoing at present. Additional discussion of thermal mechanical effects is given in *Features, Events, and Processes in Thermal Hydrology and Coupled Processes* (CRWMS M&O 2000 [142895], Sections 6.2.21 to 6.2.22) and *Features, Events, and Processes in UZ Flow and Transport* (CRWMS M&O 2000 [142945], Sections 6.8.14 and 6.8.15).

A fully coupled THC model has been developed in order to evaluate the changes in hydrologic properties caused by mineral dissolution and precipitation, and also to provide information on the thermally driven evolution of the water chemistry and gas composition in the near-field host rock over time (CRWMS M&O 2000 [153178], Section 3.3; CRWMS M&O 2000 [145774], Section 3.10). In addition to the thermal and hydrologic processes discussed in the previous section, transport of aqueous and gaseous species, mineralogical characteristics and changes, and aqueous and gaseous chemistry must also be considered. Two versions of the THC model were developed—one with all the species and minerals listed in Table 3.3-5 and a simplified version that includes fewer minerals and species. Use of two sets of minerals allows some evaluation of the sensitivity to the mineral assemblage (CRWMS M&O 2000 [153178], Section 3.3.2.4). The most important gaseous species (and the only one tracked in the model aside from water vapor) is carbon dioxide, which is closely linked to the pH of the water. Exsolution and transport of CO_2 out of the boiling zone results in a local increase in pH, and there is a corresponding decrease in pH in the condensation zone into which the vapor enriched in CO_2 is transported (CRWMS M&O 2000 [153178], Section 3.3.1.2). A number of aqueous species and mineral solids must be considered. Table 3.3-5 lists the dominant aqueous species and major mineral phases found in fractures and the matrix in the Topopah Spring welded hydrogeologic units, plus some others that are likely to occur under thermally perturbed conditions (CRWMS M&O 2000 [153178], Section 3.3.2.4).

Table 3.3-5. Species and Minerals in the Thermal-Hydrologic-Chemical Model

Aqueous Species	Gaseous Species	Minerals	
Both models:	Both models:	Both models:	
H^+	CO_2	Calcite	Amorphous Silica
Ca^{+2}	H_2O	Tridymite	Gypsum
Na^+		α–Cristobalite	Glass
H_2O		Quartz	
SiO_2		Complex model only:	
Cl^-		Hematite	K-Smectite
HCO_3^-		Fluorite	Illite
SO_4^{-2}		Goethite	Kaolinite
Complex model only:		Albite	Sepiolite
Mg^{+2}		Microcline	Stellerite
K^+		Anorthite	Heulandite
AlO_2^-		Ca-Smectite	Mordenite
$HFeO_2$		Mg-Smectite	Clinoptilolite
F^-		Na-Smectite	

Source: CRWMS M&O 2000 [142022], Tables 7 and 8

Changes in hydrologic properties are incorporated by keeping track of the volume changes resulting from mineral precipitation and dissolution. Fracture permeability changes are calculated from the porosity changes by using the cubic law, which approximates the fractures as parallel planes of uniform aperture. Matrix permeability changes are estimated using an established formula called the Carman-Kozeny relation (CRWMS M&O 2000 [153178], Section 3.3.2.6). In addition, the capillary pressure is adjusted to account for the modified aperture and pore sizes. The computed changes in fracture porosity are very small, amounting to less than one percent change everywhere in the model domain (CRWMS M&O 2000 [153178], Section 3.3.3.5). Thus, changes in hydrologic properties because of thermal-hydrologic-chemical processes are neglected in TSPA simulations (see also CRWMS M&O 2000 [142895], Section 6.2.23; CRWMS M&O 2000 [142945], Section 6.8.16). Note that the computed relative change in fracture porosity is dependent on the starting value (i.e., the fracture porosity under current conditions) of approximately 1 percent, which was based on data from gas tracer tests (CRWMS M&O 2000 [145774], Section 3.6.3.2). Also, the analysis assumes that fracture-porosity changes are uniformly distributed along fracture surfaces. The model cannot assess fine structure changes (e.g., those that reflect variation in asperities). If mineral deposition is localized in smaller openings, it may have greater impact on permeability; this possibly is still being evaluated (CRWMS M&O 2000 [153178], Section 3.3.3.5).

Information on ambient water and gas chemistry at Yucca Mountain has come from samples taken from surface-based drill holes and the ESF (CRWMS M&O 2000 [153178], Section 3.3.2.4; CRWMS M&O 2000 [145774], Section 3.8). Information on the evolution of water and gas chemistry under thermal conditions has been obtained from the thermal tests in the ESF and was used to calibrate the THC model (CRWMS M&O 2000 [153178], Section 3.6).

3.3.3.2 Implementation in the Total System Performance Assessment

The in-drift thermal hydrologic environments for TSPA simulations were computed using the multiscale thermal hydrology model (CRWMS M&O 2000 [145796], Section 3.1.4; CRWMS M&O 2000 [149862]). This model provided temperature, RH, liquid saturation and flow rate, and liquid evaporation rate at several in-drift locations. In addition, the multiscale model and the drift-scale THC model (CRWMS M&O 2000 [153178], Section 3.3; CRWMS M&O 2000 [145774], Section 3.10) were used to provide information about thermal effects on seepage. The information provided by thermal hydrology to other TSPA model components, and inputs from other TSPA components to thermal hydrology, are summarized in Figure 3.3-7.

The following sections describe the reference design as it relates to thermal hydrology, the multiscale thermal hydrology model, and the drift-scale thermal-hydrologic-chemical model and how both models are used to provide seepage volume and composition for use by the chemical environments models and other engineered-system models.

3.3.3.2.1 Potential Repository and Waste Package Design

Thermal hydrologic models require specific subsurface design and waste package design information. This information includes waste package geometry and decay-heat outputs, emplacement drift geometry and layout configuration, and waste stream information, including total thermal loading.

The reference design for the Site Recommendation is summarized in Section 1.7. The specifications relevant to this section are as follows (see also CRWMS M&O 2000 [145796], Section 2.5):

- The capacity of the potential repository is 70,000 metric tons of uranium (MTU), consisting of 63,000 MTU of CSNF and 7,000 MTU of vitrified HLW and DSNF.

- The areal mass loading is approximately 60 MTU/acre. The emplacement area encompasses 1,050 acres (4.25 km^2). (Note that only the CSNF is counted in the mass loading of 60 MTU/acre.)

- The emplacement drifts are to be ventilated for 50 years before closure. The ventilation rate is selected to remove 70 percent of the heat output before closure. The TSPA calculations assume that each waste package receives 50 years of ventilation. The potential repository would have longer ventilation durations for the initial drifts than the final drifts since emplacement takes longer than closure.

- Waste packages are placed in the emplacement drifts in a line-load configuration, with a spacing between waste packages of approximately 10 cm.

- The diameter of a waste emplacement drift is 5.5 m. Emplacement drifts are arranged with a uniform spacing of 81 m between their centerlines.

- Waste package heat output at emplacement is not to exceed 11.8 kW. The average age of the CSNF is assumed to be 26 years.

- The invert (the floor of the emplacement drifts) is composed of crushed tuff ballast in a carbon-steel-beam framework. The base-case design does not include backfill, though backfill emplaced at 50 years is considered as an alternative design (Section 4.6).

3.3.3.2.2 The Multiscale Thermal Hydrology Model

Drift-scale modeling must include coupling of drift-scale processes with mountain-scale processes to properly account for effects such as faster cooling of waste packages near the edge of the potential repository, as compared to waste packages near the potential repository center. A multiscale modeling and abstraction method was developed to couple drift-scale processes with mountain-scale processes. This method uses a series of four mountain-scale and drift-scale submodels to abstract the thermal hydrologic quantities throughout the potential repository area (Figure 3.3-8). The four submodels are described briefly in the following paragraphs. For more details, see *Multiscale Thermohydrologic Model* (CRWMS M&O 2000 [149862]).

Line-Averaged-Heat-Source, Drift-Scale, Thermal-Hydrologic Submodel–The line-averaged-heat-source, drift-scale, thermal-hydrologic (LDTH) submodel is a two-dimensional drift-scale model that computes temperature, as well as all desired moisture-related quantities (RH, liquid saturation, liquid flow rate or flux, and liquid evaporation rate) at several locations within and near the drift. This submodel effectively represents average thermal hydrologic behavior at any specific location within the potential repository, taking into account the location-specific thermal and hydrologic properties, boundary conditions, and percolation flux. Heat and fluid flow are modeled as dual-permeability, with the active fracture model for fracture flow (Section 3.3.3.1.1). Radiant heat transfer within open drifts is also modeled. This submodel addresses key issues related to modeling of fracture and matrix interactions for fluid and heat flow. It also addresses the issue of the appropriate scale of fracture properties and thermal hydrologic processes by using hydrologic properties developed for drift-scale flow by inversion models. Consistency with the mountain-scale UZ flow model ensures that hydrologic conditions are properly accounted for, including spatial variability of infiltration and changes of infiltration with climate.

Smeared-Heat-Source, Mountain-Scale, Thermal-Conduction and Smeared-Heat-Source, Drift-Scale, Thermal-Conduction Submodels–Both the smeared-heat-source, mountain-scale, thermal-conduction (SMT) submodel and the source, drift-scale, thermal-conduction (SDT) submodel consider conduction heat transfer only. They are used to establish temperature relationships and to account for the influence of potential repository edges, topography, and mountain-scale variability in hydrogeologic layering. The SMT submodel is a three-dimensional model that includes the appropriate stratigraphy, topography, and potential repository location and shape. The SDT submodel is a one-dimensional model. These two submodels address key issues related to the differences between the potential repository center and edge locations by coupling the drift-scale submodels with the mountain-scale submodel. The coupling uses the average temperatures of the potential repository host rock obtained from the SMT submodel, corrected for hydrology using a relationship derived from the LDTH and SDT submodels. The resulting modified temperatures inherently include the potential repository edge effects, because they originated from a full mountain-scale model. The temperature prediction also includes the effects of the system fluid components through the temperature relationship between the two drift-scale submodels. This results in an abstracted average drift-wall temperature that approximates the effects of the most important thermal hydrologic processes. A similar procedure is followed to abstract temperature at other locations (e.g., at the drip shield, waste package, and invert). The moisture-related quantities are abstracted by considering various correlations with each other and with temperature (see CRWMS M&O 2000 [149862], Section 6.6, for details).

Discrete-Heat-Source, Drift-Scale, Thermal-Conduction Submodel–The abstracted average quantities are further modified using waste-package-specific temperatures computed by the discrete-heat-source, drift-scale, thermal-conduction (DDT) submodel. This submodel provides information on the package-to-package heat-output variability along the drift. The DDT submodel includes only conductive heat transfer, plus radiant heat transfer within open drifts. This submodel is a three-dimensional model of a drift segment containing representative waste packages of varying heat outputs: six waste packages plus two halves (the half waste packages are on symmetry boundaries). The model drift segment contains waste package types and heat outputs representative of the overall potential repository. It consists of three 21-assembly pressurized-water-reactor CSNF packages, two 44-assembly BWR CSNF packages, one and a half codisposal waste packages, and half of a DSNF package (not co-disposal with HLW). For TSPA simulations, the DSNF was lumped together with the co-disposal waste, so the thermal hydrology results for that last half-package were not used in the TSPA. (See Section 3.5 for more information on waste types and their treatment in the TSPA.)

The multiscale thermal hydrology model involves calculations at a number of potential repository locations. A set of calculations for a given case involves only one computer run using the SMT submodel and only one computer run using the DDT submodel. However, the LDTH submodel and the SDT submodel are run for 31 locations within the potential repository and for five different areal mass loadings (to represent edge effects). The results of these 155 submodel runs are interpolated over the potential repository area and as a function of local heating conditions to provide a continuous distribution of the thermal hydrologic variables throughout the potential repository area. In order to model the preclosure ventilation period, the number of model runs is doubled, but the additional set of runs is for a shorter time period (see CRWMS M&O 2000 [149862], Section 6.6.1). Ventilation is modeled by simply reducing the waste package heat output by 70 percent for 50 years. Ventilation is also expected to dry out the drifts and the surrounding rock because dry air from the surface would be circulated through the drifts. However, this drying is neglected in the TSPA models, which is conservative in the sense that more moisture is left in the model. The abstracted temperature, RH, and other quantities are developed for an array of 610 potential repository locations (by interpolation among the 31 LDTH and SDT locations) and two waste types (CSNF and codisposal waste). The 610 locations are plotted in Figure 3.3-3.

Because resource demands for the TSPA model increase considerably as the number of environmental groups modeled within the TSPA increases, waste packages are lumped together into 30 groups for purposes of the release and EBS transport calculations (Section 3.3.2). The same thermal hydrology results are used for all three seepage states (seepage at all times, some of the time, or none of the time) in the TSPA model, so the thermal hydrology results are combined into only ten distinct groups: five infiltration bins times two waste types. However, because of the importance of the variability in waste package failure time, the full suite of 1,220 sets of results (610 locations times two waste types) are passed to the waste package and drip shield degradation models. The thermal hydrologic information passed to the various EBS models is summarized in Figure 3.3-7. (It is noted in the figure which quantities are provided as bin averages. Where bin averaging is not noted, the full set of spatial locations is used.)

The procedures used for averaging the various thermal hydrologic quantities over the infiltration bins are described in detail in *Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204]). There is one instance where the thermal hydrology abstraction does not simply average information from the multiscale model—the evaporation rate of water at the top of the drip shield. As discussed in Section 3.3.4.5.1, one of the parameters used by the chemical environments abstraction is the ratio of water evaporation rate to water flow rate (R^{es}). The multiscale thermal hydrology model can calculate these quantities only in a porous medium. When backfill is present (an alternative design), the evaporation rate and flow rate at the top of the drip shield can be provided, but with no backfill (the reference design), they must be estimated by other means. In that case, the liquid flow rate at the drip shield is taken from the seepage abstraction (Section 3.2.4), and the evaporation rate provided is a bounding estimate of how much water could be evaporated by the waste-package heat output at any given time (CRWMS M&O 2000 [152204], Section 6.3.10).

3.3.3.2.3 Coupled Effects on Seepage

The thermal perturbation caused by the heat output from the waste packages will affect seepage into emplacement drifts, in terms of both the amount of seepage and the chemical composition of the seeping water. The thermal hydrologic history of the potential repository can be divided into three stages: the early heating period, the heating period, and the cooling period (see Figure 3.3-4).

During the early heating period, drainage of thermally mobilized water leads to relatively high water fluxes above the drifts, some of which may seep into the drifts and lead to a brief period of relatively high seepage. During the heating period, most potential repository locations have temperatures above boiling and reduced saturation for some distance into the host rock. During this period, seepage may be suppressed; however, the seepage abstraction model does not take credit for this effect. Finally, there is the return to ambient temperatures during the cooling period, which takes tens of thousands of years. However, the thermal perturbation to flow around the drift is usually small after a few hundred years. Overlaid on this progression is the sequence of climate states (Section 3.2.1), so that flow and seepage do not return to their present-day conditions, but, rather, to wetter, glacial-transition conditions. Also, potentially overlaid on this progression are changes to hydrologic properties in the surrounding rock caused by coupled processes. Note, however, that such hydrologic-property changes are neglected in current TSPA simulations (Section 3.3.3.1.2).

These thermal effects on the amount of seepage are taken into account by using thermally perturbed percolation flux above the drifts from the multiscale thermal hydrology model as the input to the seepage abstraction model (Section 3.2.4.1). The percolation flux 5 m above the drifts was chosen as the quantity to use in calculating seepage. Five meters is far enough away from the drifts that the flow there is not significantly perturbed by the drift capillary barrier effect, and also far enough away that it is not usually within the boiling dry-out zone. Because of the latter, water seeps into the drifts throughout the heating period in the TSPA model (without a dry-out period). During the heating period, seepage could be reduced by evaporation or imbibition of water in the hot, dry rock in the 5 m (16 ft) above the drift. However because of the uncertainty associated with how much flow might potentially penetrate through a thin boiling zone (see, e.g., Phillips 1996 [152005]), such reduction in seepage was prevented in the TSPA model by not allowing the seep flow rate at any location to go below its initial value (i.e., the value calculated for present-day conditions).

Thermal hydrologic processes also cause changes in the chemistry of water outside the drifts and seeping into the drifts. The drift-scale THC model (CRWMS M&O 2000 [153178], Section 3.3; CRWMS M&O 2000 [145774], Section 3.10) is used to provide thermally perturbed water chemistry of seepage, which is used as a boundary condition for calculation of in-drift chemical environments (Section 3.3.4). This model computes flow of liquid and gas; transport of heat, CO_2, and numerous aqueous species (see Table 3.3-5); as well as the important chemical reactions between gaseous and aqueous species and solid minerals. Because the THC model has homogeneous hydrologic properties in the host rock around the drift, water does not seep into the drift in the model simulations. Thus, the chemistry of water in fractures just outside the drift is used to represent the chemistry of seeping water. During the period when the model has a dry-out zone around the drift, the seepage water chemistry is taken from the zone of highest liquid saturation (in fractures) above the dry-out region, which is assumed to be representative of the water that could possibly flow rapidly through fractures to the drift during the boiling period (CRWMS M&O 2000 [153178], Section 3.4.2.1).

The THC model does not account for spatial variability. It is a two-dimensional chimney model with symmetry boundary conditions, so it is most representative of the center of the potential repository, where conditions are more symmetric than they are at the edge of the potential repository. The results of this model are used in TSPA simulations to represent the water and gas composition that enters the emplacement drifts. The same compositions are used for all locations in the potential repository. It should be noted that the THC model computations were performed for an earlier version of the potential repository design. The main differences are that in the earlier design, the emplacement drifts were backfilled with quartz sand at closure and only 50 percent of the heat was removed by ventilation during the 50-year preclosure period. The addition of backfill has a large effect inside the drift, but only a minor effect on conditions outside the drift. Thus, the THC results with backfill are not expected to be significantly different from the results that would be obtained for the reference design.

3.3.3.3 Treatment of Uncertainty and Variability

The treatment of spatial and temporal variability in the thermal hydrologic environments for TSPA simulations was described, along with the rest of the implementation, in Section 3.3.3.2. A brief summary of the treatment of uncertainty and variability is that the in-drift thermal hydrologic quantities are computed for two waste types and 610 spatial locations around the potential repository. The waste package and drip shield models make use of that spatial information, but the other EBS models use information that is averaged over five infiltration bins (plus the two waste types). This averaging leaves the thermal hydrologic quantities with some spatial variability, primarily related to the variability of infiltration. The percolation flux above the drifts, which is used for calculating seepage, is also computed for two waste types and 610 spatial locations, but the seepage results are averaged over 30 environmental groups for use by the EBS models. The incoming water and gas chemistry is only computed for one representative spatial location, so it has no spatial variability in the TSPA. Both thermal models include temporal variability caused by several factors: the variability of waste package heat output over time, the active ventilation of emplacement drifts for the 50-year preclosure period, and the changes in climate at 600 years and 2,000 years, plus all the processes that govern the movement of fluids and heat (and chemical species).

The major uncertainties for thermal hydrology have to do with water and gas flow, and the thermal hydrologic calculations use the same uncertain parameters as the mountain-scale UZ flow (Section 3.2.3.3). Thus, the only uncertainty explicitly included in the TSPA is the uncertainty in infiltration rate, represented by the use of three infiltration cases: low, medium, and high.

3.3.3.4 Results and Interpretation

3.3.3.4.1 The Multiscale Thermal Hydrology Model

The amount of information generated by the multiscale thermal hydrology model is too great to show here. A few results are shown for illustration. Many more plots of results are available in *Multiscale Thermohydrologic Model* (CRWMS M&O 2000 [149862]) and *Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204]).

Figure 3.3-9 shows the computed average CSNF waste package temperature for the five infiltration bins in the medium-infiltration case. The large temperature increase occurs at the potential repository closure (50 years after emplacement), when the ventilation is turned off. During cool down, waste packages near the potential repository edge cool faster than waste packages in the center of the potential repository. The waste packages in the 0 to 3 and 60+ mm/yr (0 to 0.1 and 2.4+ in./yr) infiltration bins cool faster than the others, because all locations in those bins happen to be at the edges of the potential repository (see Figure 3.3-3). The differences among the infiltration bins is greatest during the early cooling period. By 10,000 years, they have little difference in temperature. The temperatures return to approximately ambient within 100,000 years. The results for codisposal waste packages are quite similar to those for CSNF packages; co-disposal waste temperatures are slightly lower than for CSNF packages because codisposal waste heat output is lower (CRWMS M&O 2000 [152204], Section 6.3.2).

Figure 3.3-10 shows the computed average CSNF waste package RH for the five infiltration bins in the medium-infiltration case. RH goes down sharply when the temperature rises after ventilation is turned off. Note, however, that the preclosure RH is artificially high in Figure 3.3-10, because removal of moisture by ventilation is not included in the model. As with the temperature results, the differences among the infiltration bins is greatest during the early cooling period. By 10,000 years, the bin-averaged relative humidities are above 90 percent and all within about 2 percent of each other; and by 100,000 years, they are essentially back to ambient (nearly 100 percent RH). Corresponding to their lower temperatures, the co-disposal waste package relative humidities are somewhat higher than the CSNF waste package relative humidities (CRWMS M&O 2000 [152204], Section 6.3.3).

Figure 3.3-11 shows the computed average percolation flux 5 m above the emplacement drifts for the five infiltration bins in the medium-infiltration case. As expected, the percolation flux is higher for the higher-infiltration bins. The climate changes are clearly visible at 600 years and 2,000 years. All five curves have a large spike at about 70 years, from the drainage of thermally mobilized water. The increased percolation flux can be expected to lead to a pulse of seepage into the drifts at that time. For most bins, the percolation flux in the figure does not decrease below the initial percolation flux, indicating that the boiling front usually does not reach to 5 m above the drifts.

3.3.3.4.2 The Drift-Scale Thermal-Hydrologic-Chemical Model

The results of the THC model that are of interest here are those that provide the water and gas chemistry for seepage into drifts. As discussed in Section 3.3.3.2.3, the water chemistry is taken from the fracture continuum of the model immediately above the drift. If the fractures at that location are dry at a given time, the water chemistry from the location above it with the highest liquid saturation is used. Gas composition is always taken from the location immediately above the drift.

The results of the THC model were simplified for input into the TSPA model. The simulation time was divided into four periods: preclosure (0 to 50 years), boiling (50 to 1,000 years), transitional cool down (1,000 to 2,000 years), and extended cool down (2,000+ years). For each period, representative values for CO_2 and the major aqueous species were taken from the THC model (CRWMS M&O 2000 [153178], Section 3.4.2.1). The resulting abstraction of the chemistry of fluids entering emplacement drifts is given in Table 3.3-6.

As indicated in Table 3.3-6, concentrations of most aqueous species were taken from the simplified THC model (see Table 3.3-5 and accompanying discussion). Species that were not included in the simplified model were taken from the complex THC model. The use of the simplified THC model results for the abstraction is based on the fact that it reproduces more accurately the observed changes to water and gas compositions in the drift-scale heater test (CRWMS M&O 2000 [153178], Sections 3.3.5, 3.4.2.1). Note that the additional constituents taken from the results of the more complex chemical system are, in general, trace constituents, compared to those taken from the simplified chemical system.

Table 3.3-6. Abstraction of Gas and Water Chemistry Entering Drifts

Period	Period 1: Preclosure	Period 2: Boiling	Period 3: Transitional Cool Down	Period 4: Extended Cool Down
Time	0–50 years	50–1,000 years	1,000–2,000 years	2,000+ years
Constituents from simplified THC model				
CO_2, log vol. frac.	–2.8	–6.5	–3.0	–2.0
pH	8.2	8.1	7.8	7.3
Ca^{+2}, molal	1.7×10^{-3}	6.4×10^{-4}	1.0×10^{-3}	1.8×10^{-3}
Na^{+}, molal	3.0×10^{-3}	1.4×10^{-3}	2.6×10^{-3}	2.6×10^{-3}
SiO_2, molal	1.5×10^{-3}	1.5×10^{-3}	2.1×10^{-3}	1.2×10^{-3}
Cl^{-}, molal	3.7×10^{-3}	1.8×10^{-3}	3.2×10^{-3}	3.3×10^{-3}
HCO_3^{-}, molal	1.3×10^{-3}	1.9×10^{-4}	3.0×10^{-4}	2.1×10^{-3}
SO_4^{-2}, molal	1.3×10^{-3}	6.6×10^{-4}	1.2×10^{-3}	1.2×10^{-3}
Additional constituents from complex THC model				
Mg^{+2}, molal	4.0×10^{-6}	3.2×10^{-7}	1.6×10^{-6}	7.8×10^{-6}
K^{+}, molal	5.5×10^{-5}	8.5×10^{-5}	3.1×10^{-4}	1.0×10^{-4}
AlO_2^{-}, molal	1.0×10^{-10}	2.7×10^{-7}	6.8×10^{-8}	2.0×10^{-9}
$HFeO_2$, molal	1.1×10^{-10}	7.9×10^{-10}	4.1×10^{-10}	2.4×10^{-11}
F^{-}, molal	5.0×10^{-5}	2.5×10^{-5}	4.5×10^{-5}	4.5×10^{-5}

Source: CRWMS M&O 2000 [153178], Table 3-11

3.3.4 Chemical Environments

The source term for transport of radionuclides from the potential repository in UZ and SZ water flow is radionuclide flux from inside the drifts to the UZ of the host rock. The radionuclide flux is influenced by in-drift EBS chemical environments and quantified with aqueous-phase transport models. EBS chemical environments and aqueous-phase transport occur within and at the walls of potential repository drifts where waste packages are to be emplaced. They are influenced by the seepage rate of water into the drifts, the seepage water composition, in-drift thermal and hydrologic conditions, and rock fall due to drift degradation. The seepage rate presented in Section 3.2 includes the influences of climate, infiltration rate of water into the mountain, and mountain-scale flow processes. Information on the seepage water composition and in-drift thermal and hydrologic conditions is given in Section 3.3.2 and Section 3.3.3. Rock fall due to drift degradation is treated in Section 3.3.1. Transport of radionuclides to the UZ from within the drifts, treated in Section 3.6, is the source term for radionuclide transport in UZ water flow. The UZ water flow model is treated in Section 3.2, and UZ transport is treated in Section 3.7.

This section describes the changing composition of gas, water, colloids, and solids within the emplacement drifts under the perturbed conditions of the repository. The chemical environments model serves as the vehicle for that description. It integrates and relates processes and results from several submodels that provide detailed results and interpretations.

All processes that were recognized to have potential influence on the chemical environments are described. However, not all of them were implemented quantitatively in the TSPA calculations, because some were found to have negligible effects on the calculated results. Implementation in the TSPA is discussed in Section 3.3.4.3.

The major composition changes are caused by the thermal loading of the system and the emplacement of large masses of materials that can react with water and gas in the system. Because the system will heat and then cool it will continually change. Thermal perturbation will affect the composition and movement of gas and water through the unsaturated system (UZ as described in Sections 3.2 and 3.3.3). These fluids of altered composition may enter the drifts and react with the materials emplaced during repository construction. Most of these emplaced materials will be very different in chemical composition from the host rock. In the drifts, reactions with emplaced materials such as metals and cement may alter water and gas compositions before they react with the waste package and the waste forms, and along the flow paths for radionuclide transport to the UZ. The emplaced materials may also provide additional sources of colloids that can transport radionuclides.

A view of general EBS design features, initial water movement, and potential rockfall within the emplacement drifts is given in Figure 3.3-12. Waste forms are contained in metal waste packages. These packages lie on pallets that rest on a flat invert composed of crushed host rock and metal beams. A titanium alloy drip shield, resting on the invert, covers the waste package. Drip shields are intended to divert entering water from contact with waste packages. Eventually, some drip shields may degrade and develop gaps. If it is breached, a drip shield only partially shields the waste packages from seepage water. Waste packages may be breached by corrosion leading to degradation of waste forms and release of radionuclides to the invert and the UZ of the host rock. Flow and transport paths are discussed further in Section 3.3.4.1. See Section 3.6 for a description of breaching and water flow through the drip shield.

The most direct way the EBS chemical environments may impact long-term performance is by changing the EBS components that inhibit the supply and limit the release rate of radionuclides to the near-field rock (to the UZ for radionuclide transport, see Sections 3.6 and 3.7). Chemical changes to the in-drift environment may affect the amounts and types of mobile radionuclides and the properties of the solids through which they are transported. The EBS chemical environments are sets of composition conditions under which the waste package corrodes, the waste forms degrade, the radionuclides are mobilized from the waste form, and the radionuclides migrate through the engineered barriers. In addition to these primary effects within the EBS, perturbed fluids generated in the EBS chemical environments could react with the host rock. Such alteration may change the flow pathways in the near-field rock and change conditions for UZ transport.

Several processes and interactions among in-drift gas, water, and EBS components potentially affect the in-drift chemical environments that are relevant to performance assessment. The chemical environments model integrates and relates several submodels with abstractions, described in Section 3.3.4.5, that provide results and interpretations for the following processes and interactions:

- Water and cement interactions
- Gas and water interactions
- Evaporation of water and condensation of water vapor

- Salts precipitation and salts dissolution
- Microbial activity and effects
- Corrosion and degradation of EBS components
- Water and invert interactions
- Water and colloids interactions.

The thermal hydrologic environments that influence the chemical environments in the drifts and provide inputs to the chemical environments models are described in Thermal Hydrologic Environments, Section 3.3.3. Those thermal hydrologic environments are abstracted from two sets of models, the thermal hydrology models that affect the in-drift thermodynamic environment, and models of the THC processes that affect water chemistry and gas-phase composition adjacent to the drift wall in the near field host rock. THC processes are described in Section 3.2 and Section 3.3.3.4.2. Inputs from the thermal hydrologic environments models to the chemical environments models are as follows:

- Incoming water composition
- Incoming water flow rate
- Incoming gas composition
- Temperatures in-drifts
- RH values in drifts
- Water evaporation rates in drifts.

Flow and transport paths among the EBS components define the locations and sequences for these processes, interactions, and the required thermal hydrologic inputs. These flow and transport paths are described briefly below. A detailed description is given in Section 3.6, EBS Transport.

3.3.4.1 Flow and Transport Paths and Sequences among Engineered Barrier System Components with Summaries of Associated Chemical Processes that Were Considered

The important potential flow and transport pathways among EBS components are shown with arrows in Figure 3.3-13, adapted from Section 3.6. The locations identified in Figure 3.3-13 are where the considered in-drift processes and interactions may occur. Those locations are where water compositions may change (to concentrated brines in some instances) due to evaporation, precipitation, redissolution, mixing, dilution, reactions with ambient gases, interactions with colloids, microbial activity, and other chemical reactions such as corrosion and interactions with EBS materials. In addition, barriers are breached, radionuclides are mobilized, radionuclide species are formed, and water streams are diverted and may mix at some of those locations.

The following sequence of events and processes may occur along the pathways and at locations that have been described. The events, processes, and sequences that actually occur at a particular location can be different from those occurring at other locations. For example, some packages may be breached, but never contacted by water seepage. For them, diffusion rather than advection is the dominant transport mechanism.

When conditions for seepage allow it, seepage water may enter the drift by gravity (by dripping). The water composition may have been chemically altered by reactions with cementitious grout around ground support components and with gases in the host rock. Water drips through the air gap above the drip shield where further reactions with in-drift gases can occur. The air gap is in Location 1. The surface of the drip shield, Location 2, is where water is diverted and where water evaporation, salt precipitation, brine formation, and drip shield breaching may occur. As long as it is intact, the drip shield diverts water fluxes to the invert and UZ. If the drip shield is breached, seepage water can pass through breaches in the drip shield and contact the surface of the waste package at Location 3, where it is diverted to the invert until the waste package is breached. That portion of water that passes through breaches in the waste package will contact the waste forms at Location 4. Water may also condense on waste forms at Location 4. Water advection (and diffusion when advection does not dominate) transport radionuclide-containing species from waste forms inside the waste package into and through the invert at Location 5. Three transport paths may pass through the invert to the UZ of the host rock. They come directly from the waste form, from diversion by the drip shield, and from diversion by the waste package. A fourth transport path, flow of some seepage down the drift wall to the invert or dripping directly on part of the invert outside the drip shield, is conservatively ignored in the model by assuming that all seepage drips onto the drip shield.

3.3.4.2 Integrating Conceptual Model for Features and Processes that Were Considered in Determining Engineered Barrier System Chemical Environments

The EBS chemical environments model described here is an integrating conceptual model of in-drift features and processes that were considered for predicting EBS chemical environments in the TSPA calculations. To that end, it integrates and relates concepts and results from several submodel abstractions that provide detailed results and interpretations. All of the features and processes are not of equal importance. Analyses may show that some of them have little impact on performance. A conceptual picture of the EBS Chemical Environments Model with inputs from TH environments is shown in Figure 3.3-14.

The following subsections describe the considered features and processes in a logical sequence, with cited references to relevant chemical environments sub-models, analyses, and abstractions. Locations refer to the locations in Figure 3.3-13. The EBS process model is given in *Engineered Barrier System: Physical and Chemical Environment Model* (CRWMS M&O 2000 [135097]). Summaries of results, interpretations, abstractions, and conclusions about the relative significance of the features and processes are presented in Section 3.3.4.5.

3.3.4.2.1 Temperature Increases, Then Decreases, and Water with Altered Composition Approaches and Enters Drifts (Location 1 in Figure 3.3-13)

When conditions for seepage allow it, water begins to seep into the drifts from the host rock. The conditions for seepage into the drifts is influenced by the thermal perturbation as determined by the abstracted seepage model described in Section 3.3.3. Several properties and processes in the host rock may influence the composition of water approaching the drift. They are temperature, mineral content, evaporation, precipitation, dissolution, condensation of water vapor, and gas phase composition. Those processes in the host rock are described in Sections 3.3.1, 3.3.2, and 3.3.3. Reactions of cement grout and metallic ground control components with water and gases may also alter the composition of the approaching water. After water enters, reactions with in-drift gases may alter its composition at any location in the drift. Gas phase composition may be influenced by corrosion and other interactions among in-drift materials (CRWMS M&O 1999 [125130]; CRWMS M&O 2000 [129281]; CRWMS M&O 2000 [129278]; CRWMS M&O 2000 [127818]).

3.3.4.2.2 Salts May Precipitate on the Drip Shield and Brine May Be Formed as the Drip Shield Diverts Water from the Waste Package (Location 2 in Figure 3.3-13)

Salts may precipitate on the drip shield due to evaporation of water faster than it can be replaced by seepage. Later, after some cooling occurs, water brines form by redissolution of those salts. Water brines may also form without seepage by reaction of condensed water or water vapor with dust, and their compositions are dependent on equilibration with carbon dioxide and solubilities of nitrate salts and bounding solubility values. Knowledge of water compositions on the drip shield is required to predict drip shield corrosion (CRWMS M&O 2000 [127818]).

3.3.4.2.3 Gaps May Form in Drip Shields Allowing Seepage Water to Contact the Waste Packages (Locations 2 and 3)

Gaps may form in the drip shields due to mechanical displacements of the invert or corrosion. Some water may pass through those gaps and contact the waste packages. (See Section 3.6 for a description of breaching and water flow through the drip shield.) Some or all of the water that reaches the waste packages may evaporate, as described in Section 3.3.3, and salts may precipitate on the waste packages. If the rate of water inflow exceeds the evaporation rate, the waste packages may divert some of the water directly to the invert (CRWMS M&O 2000 [127818]).

3.3.4.2.4 After Drift Temperatures Decrease, Water Vapor May React with Deliquescent Salts To Produce Brines (Locations 1 to 3 and 5)

As in-drift temperatures decrease and RH increases, water vapor may condense around and hydrate previously precipitated deliquescent salts on the drip shields, on the waste packages, and possibly in the invert. The result is formation of brines at those locations. Water vapor also may condense on deliquescent dust that could be present on waste packages, creating brine films (CRWMS M&O 2000 [127818]).

3.3.4.2.5 Reflux of Water May Occur inside the Drip Shields (under Location 3)

Water vapor may evaporate from the invert and condense under the drip shield. However, there are calculated results (CRWMS M&O 2000 [131150], Section 6.5) showing at some conditions, the RH beneath the drip shield did not reach the value of 100 percent required for condensation. It may be shown later that condensation leading to reflux can be discounted. If reflux does occur, some of the condensate may drip onto the waste package and back into the invert. Thus, some brine may be created where salts have precipitated on the waste package, and brine compositions in the invert may be changed by the refluxing water. See Section 3.6 for treatment of possible water reflux under the drip shield.

3.3.4.2.6 Drip Shields and Waste Packages May Corrode (Locations 2 and 3)

The major potential corrosive processes are stress corrosion cracking in the welded lids of the waste package and general corrosion for both the drip shield and waste package. Section 3.4 describes degradation of the waste package and the drip shield. The composition of brine in contact with the waste package would be altered by reaction with metals (corrosion) (CRWMS M&O 1999 [125130]).

3.3.4.2.7 Microbiological Activity May Affect Corrosion Rates and Drift Gas Compositions (All Locations)

Bacteria may multiply in brine, where they can cause microbially induced corrosion of the waste package (See Section 3.4.) and serve as a source term for microbial colloids in the in-drift colloid modeling. Microbiological activity may also alter in-drift gas compositions (CRWMS M&O 2000 [129279]; CRWMS M&O 2000 [129280]).

3.3.4.2.8 Water Vapor Condensation and Seepage of Water into Drifts May Dilute Brines (Locations 1 to 3 and 5)

Continued water vapor condensation and seepage would eventually dilute brines on drip shields, on waste packages, and in the invert (CRWMS M&O 2000 [127818]).

3.3.4.2.9 Corrosion May Penetrate Waste Packages and Brines or Waters May Contact and Degrade Waste Forms (Locations 3 and 4)

Stress corrosion cracks and general corrosion may penetrate the packages, allowing brine and water to contact and degrade waste forms. See Section 3.4 for treatment of waste package degradation, and Section 3.5 for waste form degradation. Compositions of water in contact with the waste forms may be determined by reactions with previously precipitated salts formed by evaporation, with waste package metals altered by corrosion, and potentially with changing in-drift gas composition. Some of the water that has passed through gaps in the drip shield bypasses (or is diverted by) the waste package and may eventually reach the invert. Some water may pass through corrosion gaps in the waste package and contact the waste form (CRWMS M&O 1999 [125130]; CRWMS M&O 2000 [127818]; CRWMS M&O 2000 [129278]).

3.3.4.2.10 Dissolved and Colloidal Materials Enter the Invert from Waste Forms by Water Advection, or by Diffusion where Advection Does Not Dominate Transport (Locations 4 and 5)

Composition of liquid water leaving the waste form through breaches in the waste package is influenced by waste form degradation and radionuclide mobilization. See Section 3.5 for treatment of waste form degradation.

In cases of significant transport by diffusion (where advective transport does not dominate), molecular diffusion is expected to be faster than diffusion of colloidal particles, and diffusion of colloids may be shown to be insignificant to performance. See Section 3.6 for treatment of EBS transport (CRWMS M&O 2000 [129280]).

3.3.4.2.11 Water in the Invert May Mix, and Dissolved and Colloidal Materials May Enter the UZ by Water Advection (and Potentially Diffusion if Advection Does Not Dominate Transport) (Location 5)

Water may enter the invert via (See Section 3.6) film flow around the inside of the drift wall, dripping directly on the portion of the invert that is outside the drip shield, diversion by the drip shield, seepage through the drip shield and diversion along the inside of the drip shield or the outside of the waste package, or directly from inside the waste package after contact with the waste form. The first two paths (film flow on the drift wall and direct dripping from the drift walls to the invert are conservatively ignored in the model by assuming all seepage drips on the drip shield. The potential for capillary flow in the invert from the UZ is included in the multi-scale thermal hydrology model.

Colloid concentrations in the invert may be limited by colloid stability in the invert water composition. Water streams entering the invert may mix and may interact with the invert materials. Water leaving the invert enters the host rock. See Section 3.6 for treatment of EBS transport (CRWMS M&O 2000 [127818]; CRWMS M&O 2000 [129283]; CRWMS M&O 2000 [129280]).

3.3.4.3 Integration of EBS Chemical Environments within TSPA

TSPA calculations use thermal hydrologic input parameters and chemical environment models with associated abstracted parameters to quantify water composition for application at the following three locations:

- Outer surface of the drip shield
- Outer surface of the waste package
- In the invert.

A colloid stability submodel among the chemical environments models is also applied to colloids in the invert that may transport radionuclides to and through the UZ.

Figure 3.3-13 shows a schematic diagram of those locations.

Water compositions on the drip shield, on the waste package (after the drip shield is breached), and in the invert are determined from lookup tables given in *Precipitates/Salts Model Results for THC Abstraction* (CRWMS M&O 2000 [151708]). The tables provide water composition values for specific input values of carbon dioxide fugacity, temperature, RH, and the ratio of water evaporation flux to incoming water flux. Input values are obtained as described in Section 3.3.3, Thermal Hydrologic Environments. Effects on water composition of water and cement interactions, water and invert interactions, corrosion of EBS components, and microbial activity have been considered and found to be negligible. Their effects will be discussed in Section 3.3.4.5.

The lookup tables were generated using models and methods described in *In-Drifts Precipitates/ Salts Analysis* (CRWMS M&O 2000 [127818]) The microbial mass may be calculated if needed with a software code MING V1.0 (CRWMS M&O 1998 [145225]) that is given in *In-Drift Microbial Communities* (CRWMS M&O 2000 [129279], Section 6.4).

Quantitative colloid stability abstraction information is provided in *In-Drift Colloids and Concentration* (CRWMS M&O 2000 [129280]) for application in the invert.

3.3.4.4 Treatment of Uncertainty

Generally, uncertainty has been treated by incorporating conservatism in choices of models and model inputs, and by choosing bounding models and parameter values. Uncertainty in the EBS chemical environments model derives from uncertainties treated in Section 3.3.3 for inputs from the TH environments and from uncertainties treated in Section 3.3.4.5 for EBS quantitative chemical environment submodels.

3.3.4.5 Submodel Results and Interpretations

Models, analyses, and abstractions described in the subsections below show that only the following processes and interactions affect the EBS chemical environments and potential radionuclide transport appreciably:

- Salts precipitation and salts dissolution
- Microbial activity and effects
- Water-colloids interactions.

Other processes and interactions have been considered and found to be negligible for the purpose of TSPA calculations.

3.3.4.5.1 In-Drift Evaporation and Precipitation

A normative precipitates and salts process model, in *Engineered Barrier System: Physical & Chemical Environment Model* (CRWMS M&O 2000 [135097], Section 6.5), was used as a validation basis for a TSPA evaporation-precipitates-salts model and abstraction. That model and abstraction were first described in *In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818]) and expanded in *Precipitates/Salts Model Results for THC Abstraction* (CRWMS M&O 2000 [151708]).

The *In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818]) was performed using the EQ3/6 code (CRWMS M&O 1998 [102837]), where possible. To extend the range of the EQ3/6 high RH model from a calculated true ionic strength of around 1 molal to approximately 10 molal, a Pitzer database was developed and used. The Pitzer database increases the concentration range of the EQ3/6 high RH model (CRWMS M&O 1998 [102837]) by about one order of magnitude. However, the model is still limited to high RH because under equilibrium conditions the mole fraction of water in solution is directly controlled by the RH. An ionic strength of 10 molal implies a mole fraction of water around 0.85. Thus, the EQ3/6 Pitzer high RH model (CRWMS M&O 1998 [102837]) is only used for RH values greater than 85 percent. An additional model, called the low RH salts model, was developed to approximate the water chemistry for RH values less than 85 percent.

The low RH model is a simple conservative and interpolative model. When the drift temperature is high and the RH is below 50 percent, the model is designed to vaporize all water and precipitate all incoming dissolved solids. As the drift cools, the RH rises. When it rises above 50 percent, all accumulated nitrate salts deliquesce and incoming nitrates remain dissolved and are allowed to concentrate to the solubility limit. This occurs because nitrate is highly soluble, and nitrate salts which are hygroscopic, are assumed to be unstable above a RH of 50 percent. Thus, at 50 percent RH and higher, the modeled location is wet. This reasonably conservative assumption reduces the duration of the estimated dry period, which is important for aqueous corrosion and transport models. For further discussions of these models, refer to *In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818]) and *Precipitates/Salts Model Results for THC Abstraction* (CRWMS M&O 2000 [151708]).

The low RH salts model is designed to make a smooth transition to the predictions of the EQ3/6 high RH model (CRWMS M&O 1998 [102837]) as the RH rises from 50 to 85 percent. The transition involves several simplifications, but it maintains mass and charge balance in the solid and liquid phases and restricts aqueous concentrations of the important soluble components to approximate effective solubilities at all times. As the RH rises from 50 to 85 percent, the liquid in the control volume transforms from a pure nitrate brine to the composition predicted by the high RH model at 85 percent. The primary mechanism for this change is the assumption that remaining precipitated soluble salts in the control volume are dissolved at exponentially increasing fractions until all soluble salts that initially precipitated in the low RH model are dissolved at 85 percent RH. While the low RH model likely oversimplifies a system that is much more complicated, it provides a starting point that at a minimum allows an approximation of an important transition from dry conditions to the concentrated solutions predicted by the EQ3/6 high RH model (CRWMS M&O 1998 [102837]).

The results of the precipitates/salts analysis are intended as input for models that couple evaporative/salts effects with other processes.

The inputs important to the Precipitates/Salts model include temperature, RH, incoming seepage rate (or flux), seepage water composition, volume fraction carbon dioxide in the gas, and steady state evaporation flux. Thus, for each combination of values for these inputs, a unique solution from the precipitates/salts model is needed.

The combinations of potential values of these inputs are unlimited. Therefore, the abstracted results from a subset of these combinations were obtained to define the response surface for interpolating results for all potential input values and combinations of input values. The resulting abstractions, in the form of response surfaces, are defined by the set of pH, chloride concentration, and ionic strength values calculated for combinations of input values. In these abstractions, the modeling period of the potential repository is divided into a set of time periods in which some of the inputs are abstracted to representative constant values. These time periods are as follows:

- Boiling period 2: 50 to 1,000 years
- Transitional cool-down period 3: 1,000 to 2,000 years
- Extended cool-down period 4: 2,000 to 100,000 years.

Seepage water composition and volume fraction carbon dioxide in the gas were abstracted to a constant set of values for each of these time periods, as described in Section 3.3.3 and given in Table 3.3-6.

These abstracted response surfaces are summarized in a set of lookup tables presented in Tables 3.3-7 through Table 3.3-9. These lookup tables include outputs from the low RH salts model (RH less than or equal to 85 percent) and the high RH salts model (RH more than 85 percent). The tables give output values of pH, molal chloride concentration, and molal ionic strength for combinations of the following inputs: RH, temperature (T), and relative evaporation rate (or flux) (R^{es}). The relative evaporation rate (or flux) refers to the ratio of the steady state evaporation flux (Q^e) to the incoming seepage rate (or flux) (Q^s).

The pH values for zero steady state evaporation flux (R^{es} less than or equal to 0) in the response surface tables are not exactly equal to the abstracted seepage pH values given in Table 3.3-6. The abstracted gas and water compositions in Table 3.3-6 were chosen to represent, with single values, compositions of incoming seepage water that change somewhat during each time period. Those abstracted compositions for Period 2 were not intended to represent exact chemical equilibrium. That is why the equilibrated pH in Table 3.3-7 for R^{es} less than or equal to 0 is different from the pH in Table 3.3-6. The slight differences in pH for the same cases in Periods 3 (Table 3.3-8) and 4 (Table 3.3-9) are likely due to differences in the thermodynamic databases used in the two models.

Table 3.3-7. Lookup Table for Period 2 - 50 to 1,000 Years

Input Parameters			**Precipitates/Salts Model Output**		
***RH* (%)**	***T* (°C)**	**$R^{es}= Q^e/Q^s$**	**pH**	**Cl (molal)**	***I* (molal)**
< 50.3%	na [a]	na	Dry	dry	Dry
50.3%	96	na	9.40	$3.71x10^{-3}$	$2.47x10^{1}$
51.0%	96	na	9.40	$5.68x10^{-3}$	$2.42x10^{1}$
53.1%	96	na	9.40	$4.09x10^{-1}$	$2.15x10^{1}$
55.2%	96	na	9.40	$6.85x10^{-1}$	$1.93x10^{1}$
60.5%	96	na	9.40	1.68	$1.15x10^{1}$
65.7%	96	na	9.40	2.40	5.89
71.0%	96	na	9.40	2.63	4.04
76.2%	96	na	9.40	2.68	3.63
81.5%	96	na	9.40	2.62	4.09
85.0%	96	na	9.40	2.55	4.69
> 85%	96	< or = 0	8.58	1.80	$6.00x10^{-3}$
> 85%	96	0.1	8.62	$2.00x10^{-3}$	$7.00x10^{-3}$
> 85%	96	0.5	8.87	$3.59x10^{-3}$	$1.20x10^{-2}$
> 85%	96	0.9	9.21	$1.80x10^{-2}$	$5.70x10^{-2}$
> 85%	96	0.99	9.28	$1.77x10^{-1}$	$3.78x10^{-1}$
> 85%	96	0.999	9.41	1.55	3.04
> 85%	96	> 0.999	9.40	2.44	4.94

DTN: MO0002SPALOO46.010 [149168]

Source: CRWMS M&O 2000 [151708]

NOTE: [a] not applicable

Table 3.3-8. Lookup Table for Period 3 - 1,000 to 2,000 Years

Input Parameters			Precipitates/Salts Model Output		
RH (%)	*T* (°C)	$R^{es} = Q^e/Q^s$	pH	Cl (molal)	*I* (molal)
< 50.3%	na[a]	na	Dry	Dry	dry
50.3%	90	na	7.64	3.73×10^{-3}	2.44×10^{-1}
51.0%	90	na	7.64	5.70×10^{-2}	2.40×10^{-1}
53.1%	90	na	7.64	4.06×10^{-1}	2.11×10^{-1}
55.2%	90	na	7.64	6.77×10^{-1}	1.89×10^{-1}
60.5%	90	na	7.64	1.63	1.10×10^{-1}
65.7%	90	na	7.64	2.28	5.65
71.0%	90	na	7.64	2.49	3.91
76.2%	90	na	7.64	2.53	3.54
81.5%	90	na	7.64	2.48	3.96
85.0%	90	na	7.64	2.41	4.53
> 85%	90	< or = 0	7.72	3.19×10^{-3}	1.03×10^{-2}
> 85%	90	0.1	7.71	3.56×10^{-3}	1.14×10^{-2}
> 85%	90	0.5	7.64	6.40×10^{-3}	1.98×10^{-2}
> 85%	90	0.9	7.45	3.20×10^{-2}	9.48×10^{-2}
> 85%	90	0.99	7.58	3.15×10^{-1}	6.60×10^{-1}
> 85%	90	0.9988	7.64	2.36	4.69
> 85%	90	> 0.9988	7.64	2.41	4.53

DTN: MO0002SPALOO46.010 [149168]

Source: CRWMS M&O 2000 [151708]

NOTE: [a] not applicable

Table 3.3-9. Lookup Table for Period 4 - 2,000 to 100,000 Years

Input Parameters			Precipitates/Salts Model Output		
RH (%)	*T* (°C)	$R^{es} = Q^e/Q^s$	pH	Cl (molal)	*I* (molal)
< 50.3%	na[a]	na	Dry	Dry	dry
50.3%	75	na	7.02	3.85×10^{-3}	2.43×10^{-1}
51.0%	75	na	7.02	5.88×10^{-2}	2.39×10^{-1}
53.1%	75	na	7.02	4.17×10^{-1}	2.09×10^{-1}
55.2%	75	na	7.02	6.93×10^{-1}	1.86×10^{-1}
60.5%	75	na	7.02	1.64	1.08×10^{-1}
65.7%	75	na	7.02	2.28	5.56
71.0%	75	na	7.02	2.49	3.87
76.2%	75	na	7.02	2.53	3.51
81.5%	75	na	7.02	2.48	3.92
85.0%	75	na	7.02	2.41	4.47
> 85%	75	< or = 0	7.19	3.30×10^{-3}	1.21×10^{-2}
> 85%	75	0.1	7.18	3.67×10^{-3}	1.32×10^{-2}
> 85%	75	0.5	7.14	6.60×10^{-3}	2.16×10^{-2}

Table 3.3-9. Lookup Table for Period 4 - 2,000 to 100,000 Years (Continued)

Input Parameters			Precipitates/Salts Model Output		
RH (%)	*T* (°C)	$R^{es} = Q^e/Q^s$	pH	Cl (molal)	*I* (molal)
> 85%	75	0.9	6.97	3.30×10^{-2}	9.85×10^{-2}
> 85%	75	0.99	7.02	3.24×10^{-1}	6.91×10^{-1}
> 85%	75	0.9988	7.02	2.41	4.75
> 85%	75	> 0.9988	7.02	2.41	4.47
> 85%	50	< or = 0	7.22	3.30×10^{-3}	1.36×10^{-2}
> 85%	50	0.1	7.22	3.67×10^{-3}	1.47×10^{-2}
> 85%	50	0.5	7.18	6.60×10^{-3}	2.31×10^{-2}
> 85%	50	0.9	7.03	3.29×10^{-2}	9.96×10^{-2}
> 85%	50	0.99	6.95	3.25×10^{-1}	7.45×10^{-1}
> 85%	50	0.9988	6.86	2.41	4.87
> 85%	50	> 0.9988	7.02	2.41	4.47
> 85%	25	< or = 0	7.05	3.30×10^{-3}	1.36×10^{-2}
> 85%	25	0.1	7.09	3.67×10^{-3}	1.51×10^{-2}
> 85%	25	0.5	7.23	6.60×10^{-3}	2.56×10^{-2}
> 85%	25	0.9	7.11	3.29×10^{-2}	1.02×10^{-1}
> 85%	25	0.99	6.99	3.25×10^{-1}	7.80×10^{-1}
> 85%	25	0.9988	6.78	2.46	5.10
> 85%	25	> 0.9988	7.02	2.41	4.47

DTN: MO0002SPALO046.010 [149168]

Source: CRWMS M&O 2000 [151708]

NOTE: [a] not applicable

As explained in *In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818]), the low RH salts model incorporates a functional relationship between RH and time. For the lookup tables, time is avoided as an independent input variable by imposing a linear relationship between RH and time. Increasing RH linearly with time from 50 to 85 percent provides the abstraction used to generate the lookup values for RH less than or equal to 85 percent.

The ionic strength values presented in the lookup tables are an approximation of the true ionic strength, as described in *In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818]). An additional approximation is required for lookup table pH values when the RH is less than or equal to 85 percent. Because pH cannot be calculated using the low RH salts model, it is approximated by using the EQ3/6 high RH model to perform a simple evaporation of the incoming seepage water to a true ionic strength of 10 molal (i.e., to a water activity of approximately 0.85). These values for pH are included in the lookup tables for cases in which RH is less than or equal to 85 percent.

Finally, for the case in which the relative evaporation rate is 1.0 or greater, the ionic strength and chlorine concentrations are set at the values obtained by the low RH model at 85 percent RH for the given carbon dioxide fugacities and temperatures. This is done to approximate a reasonable transition between the low RH and high RH model results.

Details on implementation of the precipitates-salts model and abstractions are reported in *In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818]) and (CRWMS M&O 2000 [151708]).

Uncertainties

Although simplifying assumptions were required to reduce the complexity of the precipitates/salts analysis and to avoid sophisticated approaches where data were lacking, these assumptions tended to err on the side of conservatism. In particular, they tended to result in a shorter dry period by not allowing dry conditions above a RH of 50 percent and in higher chloride concentrations at lower relative humidities. Judging by the accuracy of the model predictions compared to experimental data (CRWMS M&O 2000 [127818], Section 6.5), the greatest uncertainties of the precipitates/salts analysis for TSPA are likely the thermal hydrologic and THC predictions and other predicted inputs that feed the analysis.

A final method used to evaluate and account for uncertainty in the precipitates/salts analysis is the generation of a set of lookup tables intended to cover the range of possible combinations of input values. These lookup tables can be used in several ways. Initially, they can be used to evaluate the sensitivity of input variables on outputs. For example, the sensitivity of pH to the relative evaporation flux can be evaluated by comparing the pH output for a range of values for the relative evaporation flux. In the precipitates/salts analysis, input variables that are not included in the tables are not sensitive inputs. Similarly, an estimate of the approximate maximum range of possible values of a given output variable for a range of input conditions can be assessed from the lookup tables. However, the primary objective of the lookup tables is to summarize the effects of evaporation processes for a wide range of possible conditions so that downstream users (e.g., corrosion modelers or developers of an in-drift geochemical model abstraction) can easily incorporate evaporation effects and uncertainty into coupled analyses. Uncertainties due to spatial heterogeneities can be taken into account using the wide range of conditions covered in the lookup tables.

3.3.4.5.2 In-Drift Microbial Activity

A microbial effects process model in *Engineered Barrier System: Physical and Chemical Environment Model* (CRWMS M&O 2000 [135097], Section 6.4) has been used to develop a set of threshold conditions for microbial growth and activity. It is based on information from the literature describing the environmental conditions for which microbial growth and activity are observed. The model is bounding in the sense that extreme microbial observations (e.g., halophiles and hyperthermophiles) are included, but these types of organisms will not necessarily be important in the potential repository. No distinction is made between environmental conditions necessary for microbial activity and for biofilm development, which is conservative. This model confirms the thresholds for microbial activity in the in-drift microbial communities model described below.

An in-drift microbial communities model given in *In Drift Microbial Communities* (CRWMS M&O 2000 [129279]) was developed to bound the microbial communities that could be present within a given length of the potential repository drift. In general, the model uses constraints on the supply rates of the nutrients to build an idealized microbial composition, comprised of carbon, nitrogen, sulfur, and potassium, in addition to the water components. The rates of supply of these constituents are input as constant release rates for each introduced material in the system by specifying the mass and composition of the material and its degradation lifetime. The other major constraint evaluated is the energy available for microbes to grow based on the pH-corrected, standard state free energy released from oxidation and reduction reactions. Other constraints on microbial growth are temperature and RH thresholds in the model that limit the start of microbial activity until the boiling period is over. Although microbes could be sterilized out of the drifts during the highest temperature period, because they are present in the water-rock system they will return as water drips back into potential drifts.

The results of the conceptual model were incorporated within the MING V1.0 software (CRWMS M&O 2000 [129279]) during software development or are incorporated directly as parameter inputs. This model and the MING software are available to calculate upper bounds on in-drift microbial populations, as needed.

Microbes can accelerate corrosion and this effect has been taken into account by applying a conservative general corrosion enhancement factor. The enhancement factor is represented with uniform distribution between 1 and 2.0 (see Section 3.4.1.6).

The effect of microbial activity on in-drift gas composition is undetermined.

Uncertainties

This model is an upper bounding model for microbial activity. Results of validation tests indicate that (1) the model will function as intended, and (2) the model predictions are accurate to within one order of magnitude of measured values (CRWMS M&O 2000 [129279], Section 7.3).

3.3.4.5.3 In-Drift Water Colloids Interaction

An EBS colloids process model was developed and reported in *Engineered Barrier System: Physical and Chemical Environment Model* (CRWMS M&O 2000 [135097], Section 6.6). The model bounds the impact of iron-oxide and iron-oxyhydroxide colloids on radionuclide transport in the invert, specifically considering the impacts resulting from use of steel in the engineered barrier system.

An in-drift colloids model and an abstraction reported in *In-Drift Colloids and Concentration* (CRWMS M&O 2000 [129280]) were developed. Clays, silica, hematite, and goethite colloids occur as mineral colloids in groundwater in the vicinity of Yucca Mountain, with clays and silica the most abundant. It is assumed that these colloids will enter a failed waste package along with groundwater and be available to interact with released radionuclides. Further, it is assumed that groundwater entering the drift and invert from the surrounding UZ will mix under certain circumstances with releases from a failed waste package and the groundwater colloids will likewise be available to interact with released radionuclides.

The predominant process of colloidal radionuclide release occurs when the drip shield and waste package have been breached and incoming water from above flows downward through the breach in the drip shield, into and around the waste package, and downward into the invert.

Releases from a breached waste package may mix with the groundwater in the invert. The mixed fluid migrates downward through the invert into the UZ.

Although considered, diffusion of colloidal particles is too slow to be a significant contributor to radionuclide releases (CRWMS M&O 2000 [135097], Section 6.6).

The water is assumed to contain smectite colloids whose stability and concentration are determined by the ionic strength and pH of the groundwater. The waste package release is assumed to be a fluid containing colloids and dissolved radionuclides resulting from the reaction of waste with water that has entered the waste package (calculated by the TSPA-SR model).

There are three types of colloids in the release: (1) waste form colloids, assumed to be smectite; (2) waste package corrosion colloids, assumed to be iron (hydr)oxide; and (3) groundwater colloids, assumed to be smectite. Some of these colloids have associated radionuclides as they leave the waste package. The waste form colloids may have irreversibly attached (embedded) and/or reversibly attached (sorbed) radionuclides. The corrosion and groundwater colloids may have reversibly attached radionuclides. It should be stated that the terms "reversible" and "irreversible" as used here imply mechanism of attachment.

The mass of radionuclides irreversibly attached to the waste form colloids is determined from reactions within the waste package (CRWMS M&O 2000 [148214]). The mass of radionuclides reversibly attached to all three types of colloids is determined by the product of three parameters: (1) the mass concentration of dissolved (aqueous) radionuclide in the fluid, (2) the mass concentration of colloid material in the fluid, and (3) the distribution coefficient, k_d, of a specific radionuclide on a specific colloid mineralogical type. The k_ds for the various radionuclides on the two mineralogical colloid types have been determined in the laboratory (CRWMS M&O 2000 [129280], Section 6.3.4).

Stability and mass of waste form colloids were abstracted for TSPA calculations and are depicted in Figures 3.3-15 and 3.3-16 (CRWMS M&O 2000 [129280], Section 6.3.4.3, Figures 4 to 5)

Stability of iron (hydr)oxide colloids has been abstracted and is depicted in Figure 3.3-17 (CRWMS M&O 2000 [129280], Section 6.3.4.4, Figure 7).

Groundwater colloid concentration as a function of ionic strength has been abstracted and is depicted in Figure 3.3-18 (CRWMS M&O 2000 [129280], Section 6.3.4.5, Figure 9).

Details on implementation of the in-drift colloids model and abstractions are reported in *In-Drift Colloids and Concentration* (CRWMS M&O 2000 [129280], Section 6.3).

Uncertainties

There are several significant sources of uncertainty attached to this water-colloid interaction abstraction, some relating to assumptions regarding colloid generation from degradation. The potential formation of colloids from degradation of N-reactor fuel, and its potential contribution to potential repository performance, must be investigated. At this time, the data are preliminary; however, the program is ongoing, and more data are anticipated. For now, it is assumed in the abstraction that, due to the small quantity of N-reactor fuel, any colloids generated from degradation of the fuel will have little effect on potential repository performance. If this assumption, after examination, proves untrue, use of the assumption could result in underestimation of the contribution of colloids derived from degradation of N-reactor fuel to potential repository performance.

Another uncertainty, but less significant, is the assumed concentrations of colloids in groundwater. Data from a range of groundwaters were used in an attempt to establish a fundamental relationship between colloid concentration and ionic strength. The large amount of scatter in the data were accommodated by bounding the data. As a result, colloid concentrations in some circumstances may be overestimated.

The abstraction is considered valid and usable in TSPA calculations for any time after the temperature in the potential repository has decreased to well below boiling. Many of the waste degradation tests were performed at 90°C, but mostly sampled near room temperature. Therefore, the test results may be applied to drift processes during the post-thermal period. The range of ionic strength and pH, for which colloid masses and stability are calculated in the abstraction, are within the ranges anticipated from in-drift chemistry calculations and abstraction.

In general, the bounding relationships employed in the abstraction incorporate uncertainty present in the data used. Additional uncertainty may result from implementation in the TSPA-SR model calculations. For example, the choices of distributions and the method of sampling a particular distribution may result in uncertainties in determination of colloid concentrations, ionic strength, pH, and radionuclide concentrations.

3.3.4.5.4 Water-Cement Interactions

An EBS cementitious materials process model was developed and reported in *Engineered Barrier System: Physical and Chemical Environment Model* (CRWMS M&O 2000 [135097], Section 6.3). The model is used to develop reasonable-bound estimates for potential chemical effects from percolating water that contacts grouted rockbolts in the drifts and for interaction of that water with gas-phase CO_2. The grout has low permeability, which substantially limits chemical interaction of the grout with the EBS environment. Flux scaling produces greater flow rates than does limiting leachate flow by the saturated permeability of the grout. Even with flux scaling, the composition and quantity of leachate after equilibration with CO_2 are of minor importance compared to the composition of water in the bulk environment.

For these reasons, effects of cement leachate on the composition of water in the bulk environment can be neglected.

3.3.4.5.5 In-Drift Corrosion Products

A scoping calculation reported in *Engineered Barrier System: Physical and Chemical Environment Model* (CRWMS M&O 2000 [135097], Section 6.7.4) shows the needed oxygen is available in the host rock and can be replaced by natural processes, at a rate that is comparable to the rate of corrosion.

An in-drift corrosion products model and abstraction were developed in *In Drift Corrosion Products* (CRWMS M&O 1999 [125130]). The reported conclusion was that only minor impacts of corrosion are expected in the bulk in-drift chemical environment. These impacts may occur during active corrosion of the metals and alloys in an oxidizing environment. After formation, corrosion products are generally insoluble in an oxidizing environment and should not affect the composition of the solution further. However, there is a large potential for sorption, which has not been fully quantified nor have its impacts on the bulk in-drift geochemistry been evaluated.

It is concluded that effects of corrosion will have negligible effect on the in-drift chemical environment.

3.3.4.5.6 In-Drift Gas-Water Interactions

A gas flux and fugacity process model was developed and reported in *Engineered Barrier System: Physical and Chemical Environment Model* (CRWMS M&O 2000 [135097], Section 6.2). It is an analytical model for fugacities of CO_2 and O_2 in the potential repository during the thermal period. The model provides lower-bound estimates of gas fugacities. The results show the advective-dispersive oscillatory barometric pumping process is a potentially important mechanism for gas transport in the UZ.

An in-drift gas and water interactions analysis and abstraction were reported in *In-Drift Gas Flux and Composition* (CRWMS M&O 2000 [129278]). The analysis provides the basis for constraining the values of the gas flux and composition in the potential emplacement drifts in terms of the major geochemical constituents carbon dioxide, oxygen, nitrogen, and water vapor. The conceptual analysis and mass balance calculations suggest that in-drift gas flux and composition will not be strongly affected by interactions with in-drift and near-drift materials. However, in-drift gases will be displaced by water vapor during the thermal period, thus, dropping the levels for all other gases in the drifts during that period. Even this decrease in oxygen fugacity is not expected to be great enough to reverse redox reactions occurring in the potential repository.

It is concluded that the boundary conditions, which use in-drift gas compositions and fluxes, ignoring interactions with materials in and near the drifts, will not be significantly changed by those interactions.

3.3.4.5.7 Water-Invert Interactions

A water-invert conceptual model and an abstraction were reported in *Seepage/Invert Interactions* (CRWMS M&O 2000 [129283]). It was concluded that the invert materials are not present in significant quantities relative to the host rock and other EBS materials to exert a significant influence on the chemistry of the seepage exiting the drift. Due to the potential for mineral precipitation during the thermal period and re-solution during the return to ambient temperature conditions, it is possible that there will be a transient period of elevated ionic strength for seepage exiting the drift through the invert.

Potential changes in invert transport properties have little impact. The invert is filled with crushed tuff generated from mining operations on the emplacement tunnels. The invert is therefore expected to respond to the effects of heating and water-rock interaction during seepage and rewetting in a similar manner to the host rock. In particular, the hydrological properties are expected to be relatively constant during the processes of heating and water-rock interaction, particularly for a high porosity, granular material.

An additional factor mitigates the effects of potential changes in hydrological properties on the crushed tuff. The typical dimension of the invert, about 1 meter deep in cross section, is much less than the typical size scale for the UZ, on the order of hundreds of meters above the water table. In this situation, potential changes to invert properties will be small perturbations in comparison to the overall response of the host rock in the UZ.

3.4 WASTE PACKAGE AND DRIP SHIELD DEGRADATION

The waste package and drip shield together form the primary component of the EBS (Figure 3.4-1). The waste form will be completely contained and out of contact with groundwater until the waste package is breached. The current approach used in the analysis of waste package degradation considers the important degradation mechanisms (e.g., corrosion) that may eventually cause such a breach, or failure, to occur. Mobilization of waste within the waste package, and subsequent transport of radionuclides into the natural environment are described in Sections 3.5 and 3.6. Waste package and drip shield degradation are described in detail in the *Waste Package Degradation Process Model Report* (CRWMS M&O 2000 [151624]).

Figure 3.4-1 illustrates the waste package and drip shield design. The package is dual walled: a 20 mm thick Alloy-22 outer wall and a 50 mm thick stainless steel (316NG) inner wall (CRWMS M&O 2000 [144128], Attachment I, p. 2 of 2). (Note that the outer wall of the waste packages for the defense high-level waste and navy spent nuclear fuel is 25 mm thick [CRWMS M&O 2000 [150823], Attachment III, p. III-1). The purpose of the outer wall is to provide corrosion resistance, and that of the inner wall is to provide structural support (CRWMS M&O 2000 [144128], p. 31). The waste package is initially constructed as a cylinder with one end closed. Two lids are located at the closed end of the cylinder: a 95 mm thick stainless steel lid closing the inner wall, and a 25 mm thick Alloy-22 lid closing the outer wall. The entire assembly is then annealed to reduce residual stresses resulting from the fabrication process, which can cause stress corrosion cracking or accelerate other corrosion processes. The waste form is placed in the package, and the package is sealed by welding three closure lids onto the open end. The inner wall is closed with a single 95 mm thick stainless steel inner lid. Two Alloy-22 closure lids (referred to as the outer and middle lid, respectively) are welded onto the outer wall of the waste package. The outer lid is 25 mm thick and the middle lid is 10 mm thick (CRWMS M&O 2000 [144128], Attachment I, p. 2 of 2). Although large-scale annealing of the waste package closure welds is not possible, localized stress-relief treatments (induction annealing of the outer lid welds and laser peening of the middle lid welds) will be applied to the closure welds (CRWMS M&O 2000 [144128], Section 6.4). These treatments will result in the formation of compressive surface stresses to a depth of 2 to 6.5 mm. SCC will not initiate until these compressive regions are removed by general corrosion processes. The localized stress-relief treatments will not result in appreciable heating of the spent fuel elements within the waste package (CRWMS M&O 2000 [144128], Section 6.4). Depending upon the type of waste form, additional barriers may be present. For commercial spent nuclear fuel, a fuel rod cladding, typically a Zirconium-alloy metal, surrounds the fuel pellets. The form of the fuel itself, as uranium oxide ceramic pellets, also provides some degree of immobilization for the heavy metal components. Cladding and fuel related issues are discussed in Section 3.5.

The drip shield provides additional protection for the waste packages. Constructed of 15 mm thick Titanium Grade 7, the drip shield diverts water entering the drift from above, thus preventing seepage from contacting the waste package.

In the nominal scenario, general and localized corrosion (i.e., pitting/crevice corrosion and stress corrosion cracking) mechanisms are addressed as possible waste package and/or drip shield failure or breaching modes. General corrosion refers to corrosion processes that are spatially continuous (although variable) over the entire surface, or a substantial portion of the surface, of a single waste package or drip shield, in response to general environmental conditions in the vicinity of the waste package and drip shield. For example, humid air corrosion is a general corrosion mechanism. Localized corrosion processes effect only a small area, however, within that area they may have a greater impact on package integrity. In the current analyses, SCC may occur at the welds where the outer and middle closure lids are joined to the outer wall of the package body after waste is placed within the package. Pitting and crevice localized corrosion processes were considered in TSPA-VA (CRWMS M&O 1998 [108004]). However, the current waste package and drip shield design incorporates materials for which pitting and crevice corrosion will not occur under foreseeable repository conditions (CRWMS M&O 2000 [147648], Section 7.1).

Mechanical failure modes (e.g., due to rock fall) are not considered in the present analyses. They have been excluded due to low consequence as a result of the FEP analysis (CRWMS M&O 2000 [146538], Section 6.2.19).

Process models and abstractions have been developed to simulate the degradation of the drip shield and waste package. The models are intended to capture the spatial and temporal variability of corrosion processes by integrating predicted localized environmental conditions with submodels describing corrosion processes for a representative number of waste packages and drip shields. The models have been simplified to the greatest extent possible, while still capturing the essential behavior of the system. Where simplifications have occurred, they have been conservative in nature. For example, no corrosion credit is taken for the stainless steel inner wall of the package.

The following subsections describe construction of the conceptual models, the implementation of the models as computer codes, and discuss the results of model application.

3.4.1 Construction of the Conceptual Model

The conceptual model (CRWMS M&O 2000 [146427], Section 6.2) includes those waste package and drip shield degradation mechanisms that may occur under the predicted environmental conditions in the potential repository. A number of other degradation mechanisms were investigated, but were excluded from the final analysis as having either low probabilities of occurrence or no significant contribution to degradation, and therefore, no significant contribution to the calculated expected dose (CRWMS M&O 2000 [146538]).

The conceptual model consists of quantitative or mathematical descriptions of the progression of each modeled degradation mode for each applicable component (drip shield or waste package) (CRWMS M&O 2000 [146427], Section 6.2).

The modeled degradation mechanisms are a function of the material properties of the drip shield and waste package, and the sequence of events that is anticipated to occur subsequent to repository closure. Three main types of degradation are considered in the nominal case: humid-air general corrosion, aqueous general corrosion, and SCC. Two additional corrosion processes, microbially induced corrosion (MIC), and thermal aging and phase instability, are considered to provide enhanced general corrosion on the waste package. General corrosion mechanisms are conceptually similar for the drip shield and waste package and are simulated using a common approach (CRWMS M&O 2000 [146427], Section 6.3.5 and 6.3.6).

3.4.1.1 Model Input and Output

The primary models supplying input to the drip shield and waste package degradation abstractions are the Thermo-Hydrology Model and the In-Drift Geochemical Abstraction Model (Figure 3.4-2). Inputs to the drip shield degradation model consist of: drip shield design data, emplacement drift temperature and RH profiles as a function of time, and general corrosion rate data (Figure 3.4-3) (DTN: MO0007MWDTSP01.003 [151706]). Inputs to the waste package degradation model consist of: waste package design data, emplacement drift temperature and RH profiles as a function of time, general corrosion rate data, closure lid weld stress and stress intensity factor profiles, stress corrosion cracking model parameters, manufacturing defect prediction parameters, as well as in-package chemistry data (Figure 3.4-4) (DTN: MO0007MWDTSP01.003 [151706]).

It was anticipated that other data, such as occurrence of dripping and non-dripping conditions associated with in-drift seepage, would be required. However, by adjusting scenarios for conservative cases (e.g., it is conservatively assumed that an aggressive dripping water chemistry is present over all time), the input data requirements and associated model complexity were reduced. Other anticipated input data such as in-drift water and gas compositions and chemical properties were found to be unnecessary as several material corrosion rate properties for Alloy-22 (the waste package) and Titanium Grade 7 (the drip shield) were not sensitive to variations in these data over the expected range (CRWMS M&O 2000 [144229]; CRWMS M&O 2000 [144971]).

Output from the degradation models is a time dependent quantitative assessment of the drip shield and waste package degradation and failure. Results include: the time to initial breach for the drip shield and the waste package; time to first breach of the waste package by stress-corrosion crack failure; and the degree of drip shield and waste package failure as a function of time (see Figures 3.4-3 and 3.4-4). The time of the first breach of the waste package corresponds to the start of waste form degradation within the breached package.

Additional output data include the uncertainty and spatial variation of the degradation information for each waste package and drip shield basis and at different locations within the potential repository.

Figure 3.4-5 is a conceptual illustration of model data sources, input data used by the model, and output data generated by the model.

3.4.1.2 General Corrosion

General corrosion normally causes a relatively uniform thinning of materials. Two types of general corrosion are considered: humid-air corrosion and aqueous corrosion. Humid-air corrosion occurs when the relative humidity at the surface of the drip shield and waste package in the emplacement drift exceeds a threshold value. The threshold relative humidity used in the current analysis is based on the deliquescence point of sodium nitrate salt ($NaNO_3$) which is a function of temperature (CRWMS M&O 2000 [146460]; CRWMS M&O 2000 [144229]). While this threshold is exceeded, general corrosion will cause the material to thin according to a material dependent corrosion rate. Aqueous corrosion will occur when a material surface is wetted, as from seepage or drips. When wetted, the material will also thin according to a material dependent corrosion rate. Figure 3.4-6 illustrates general corrosion processes.

The corrosion rate is theoretically a function of a number of variables, including temperature, stress-state, and water and gas chemistry. However, laboratory testing determined that the rates of general corrosion for Alloy-22 (the waste package) and Titanium Grade 16 (an analog for the Titanium Grade 7 used for the drip shield) (CRWMS M&O 2000 [144971], Section 6.5.2) are insensitive to temperature, relative humidity, and liquid pH in the range expected for these parameters within the potential repository (CRWMS M&O 2000 [144229]; CRWMS M&O 2000 [144971]).

The corrosion rate is characterized by a probability distribution that contains both variability and uncertainty. "Uncertainty" describes the lack of knowledge concerning the exact corrosion degradation rate of a material, while "variability" refers to the differing corrosion degradation rates that could occur because of different or varying material properties (on a microstructural scale) and temporal or spatial exposure conditions. Within the model implementation, the variability and uncertainty are separated using Gaussian Variance Partitioning (see Section 3.4.2.2).

For each material, both humid-air and aqueous general corrosion use the same corrosion rate distribution because the general corrosion rates of the materials (Alloy-22 and Titanium Grade 7) do not show any significant dependence on the exposure conditions over the ranges that are expected in the potential repository (CRWMS M&O 2000 [144229]; CRWMS M&O 2000 [144971]). Conceptually, the two processes are differentiated by the possibility of localized pit corrosion initiation, which may occur under aqueous general corrosion conditions.

However, as discussed previously, neither waste package nor drip shield materials are subject to localized pitting under expected repository conditions. The following sequence describes the progression of general corrosion modes in response to system behavior and events:

- Initially, the upper surface of the drip shield is assumed to be dripped upon, and aqueous general corrosion conditions are assumed. The underside of the drip shield and the exterior of the waste package are subject to humid-air general corrosion. The interior of the waste package is not subject to general corrosion.

- After several tens of thousands of years, general corrosion penetrates the drip shield in one or more locations. Consequently, the corrosion mode for the exterior of the waste package is changed from humid-air general corrosion to aqueous general corrosion as dripping water contacts the waste package surface.

- After failure of the waste package (either by general corrosion or SCC) the interior of the waste package is subject to aqueous general corrosion under in-package water chemistry conditions (see Section 3.5.2).

3.4.1.3 Stress Corrosion Cracking

SCC is a crack propagation process caused by the combined and synergistic interaction of mechanical stress and corrosion reactions. For SCC to occur, tensile stress (stress that would tend to open a crack) and an aggressive water chemistry must be present simultaneously. It is conservatively assumed that an aggressive dripping water chemistry is present over all time. The drip shield will be fully annealed to relieve all tensile stresses resulting from the fabrication process. Therefore, the only source of mechanical stress in the drip shield is the loading due to potential rockfall. Although SCC due to rockfall induced stress states is possible, it is of low consequence because SCC cracks in the Titanium Grade 7 drip shield will be very tight and will quickly become "plugged" by corrosion products and precipitates such as carbonate present in the seepage water. Any water transport through this oxide/salt filled crack area will be mainly by diffusion-type transport processes (CRWMS M&O 2000 [147396]). Thus, the effective water flow rate through SCC cracks in the drip shield would be expected to be extremely low and will not contribute significantly to the overall radionuclide release rate from the potential repository. All the fabrication welds on the waste package, except the welds for the closure lids (see Section 3.4 for a brief discussion of the localized stress relief treatment employed), will be fully annealed and thus not subject to SCC. The waste package is protected from rockfall by the presence of the drip shield. SCC is therefore considered only for the closure-lid welds. It is assumed that SCC is operative if the relative humidity of the waste package surface is greater than the threshold relative humidity (i.e., general corrosion is occurring).

The waste package outer barrier has a dual closure lid design to mitigate potential premature failure of waste packages by stress corrosion cracking. The dual closure lids are referred to as the outer lid and middle lid. The outer lid is 25-mm thick and the middle lid is 10-mm thick. There is a physical separation between the two lids. A schematic of the dual closure lid design is shown in Figure 3.4-7.

Any SCC cracks initiated in the outer closure lid stop after the lid is fully penetrated. As mentioned above, SCC will not occur unless general corrosion has started. Consequently, SCC on the middle closure lid weld will not start until the outer lid is breached, opening the space between the outer and middle lids to potential repository conditions, and allowing general corrosion to occur on the middle-lid.

The growth of cracks due to SCC is modeled with the Slip Dissolution Model (CRWMS M&O 2000 [148375]). This model predicts the velocity of crack growth as a function of stress intensity factor, incipient crack size, and crack growth rate parameters. If the predicted stress at an incipient crack location is greater than the threshold stress and the stress intensity factor is greater than zero, then the crack can grow in depth. The crack continues to grow as long as the predicted stress at the crack tip exceeds the threshold stress and the stress intensity factor is greater than zero.

Stress and stress intensity factor profiles for each closure lid are calculated at a number (i.e., five) of angles around the closure lid to represent variability (DTN: MO0007MWDTSP01.003 [151706]). Uncertainty in the stress state is introduced through use of a fraction (of yield strength) that defines a maximum deviation from the median stress/stress intensity factor profile and a random variable to assign uncertainty in this range (CRWMS M&O 2000 [146427], Section 4.1.8). By changing the fraction, differing conceptual models (e.g., conservative 30 percent, expected 10 percent, and optimistic 5 percent) (CRWMS M&O 2000 [151624], Section 3.2.5) for the stress states can be represented. Stress and stress intensity factor profiles for the 25-mm outer lid are shown in Figures 3.4-8 and 3.4-9 for the different choices of the uncertainty fraction. Similarly, stress and stress intensity factor profiles for the 10-mm middle lid are shown in Figures 3.4-10 and 3.4-11 for the different choices of the uncertainty fraction.

SCC cracks in passive alloys such as Alloy-22 tend to have small crack opening displacements (i.e., tight cracks) by nature (CRWMS M&O 2000 [147396]). The opposing sides of through-wall SCC cracks will continue to corrode at very low passive corrosion rates until the gap region of the tight crack opening is plugged by the corrosion product particles and precipitates such as carbonate from ionic constituents present in the water. Any water transport through this oxide and salt-filled crack area will be mainly by diffusion-type transport processes (CRWMS M&O 2000 [147396]). Thus, the earliest radionuclide release from the waste packages due to diffusion through the SCC cracks would be expected to be low relative to the overall radionuclide release from the potential repository due to later advective transport. Because radiation dose is a function of the radionuclide release rate, there should be no significant change to the expected annual dose.

3.4.1.4 Manufacturing Defects

Manufacturing defects considered in the present analysis consist of undetected non-through going cracks in the closure lid welds (CRWMS M&O 2000 [147359]) as shown schematically in Figure 3.4-12. After waste is placed in the waste package, the closure lids are welded onto the open end of the package. All welds will be inspected with various non-destructive testing procedures, however, it is possible that some defects may not be detected.

The weld defects are assumed to be spatially randomly distributed as represented by a Poisson process (CRWMS M&O 2000 [146427], Section 5.5). These assumptions are reasonable for the manufacturing process being considered (CRWMS M&O 2000 [138164]). All weld defects are conservatively assumed to be oriented radially (i.e., perpendicular to the weld centerline). Thus, they propagate under the action of the hoop stress profile which is more severe than the radial stress profile. Most weld defects, such as lack of fusion and slag inclusions, would be expected to be oriented within a few degrees of the weld centerline (CRWMS M&O 2000 [151624], Section 3.1.2.6). Other parameters required for the calculation include: lid thickness, lid radius, and location and scale parameters for the probability of non-detection of defects. An additional parameter represents the fraction of propagating defects. Undetected defects serve as nucleation sites for additional cracks whose growth is then modeled by SCC subsequent to the start of general corrosion.

This approach differs from that employed in the previous performance assessment (DOE 1998 [100550], Volume 3, Section 4.1.7), where a percentage (between 0.001 and 0.1 percent) of the waste packages were assumed to fail 1,000 years after potential repository closure. The incorporation of the dual closure lid design in the current assessment effectively precludes initial failure as the probability of undetected weld defects penetrating the entire thickness of both the outer- and middle-closure lid is exceedingly remote.

3.4.1.5 Localized Corrosion

Localized corrosion, or pitting and crevice corrosion, is induced by local variations in the electrochemical potential or driving force for corrosion on a micro-scale over small regions. Initiation of localized corrosion requires aqueous general corrosion (i.e., dripping conditions) and specific chemical conditions to initiate.

It is assumed that localized corrosion of Titanium Grade 7 is not possible under all expected repository conditions (CRWMS M&O 2000 [146427], Section 5.3). Therefore, localized corrosion of the Titanium Grade 7 drip shield is not modeled. Similarly, localized corrosion is not possible on the Alloy-22 waste package under potential repository conditions, however, a localized corrosion initiation and propagation model is implemented within the WAPDEG model (CRWMS M&O 2000 [146427], Section 5.4).

3.4.1.6 Other Degradation Modes

A number of other degradation modes were considered for incorporation in the drip shield and waste package degradation analysis.

Microbially Influenced Corrosion–This is caused by the metabolic activity of microorganisms. In the Analysis Model Report entitled General Corrosion and Localized Corrosion of the Drip Shield (CRWMS M&O 2000 [144971]), it is stated that the effect of microbial growth on the corrosion potential is not significant and the initiation of crevice corrosion under bio films formed on titanium has never been observed. Thus, the drip shield material (Titanium Grade 7) is assumed not subject to MIC. It is assumed that the waste package material (Alloy-22) is subject to microbially induced corrosion only when relative humidity exceeds 90 percent and sufficient nutrients exist, and that MIC can be represented by a general corrosion enhancement factor (CRWMS M&O 2000 [144229], Sections 6.8 and 6.10). The enhancement factor is represented with uniform distribution between 1 and 2.0.

Aging and Phase Instability–Prolonged exposure to elevated temperature environments can cause microstructural changes of waste package and drip shield materials, potentially resulting in changes in their corrosion behavior such as enhanced general corrosion. The drip shield is assumed to be immune to long-term aging and phase instability. This assumption is based on the fact that Titanium Grade 7 is a nearly pure single-phase alloy with very small additions of alloying elements (such as Pd) and that the phase transition temperature for the alloy (about 880°C) is much higher than the expected peak exposure temperatures of drip shields in the potential repository (CRWMS M&O 2000 [144971], Section 5.9). The bounding analyses based on the limited data that are currently available indicate that Alloy-22 base metal will not be subject to the long-term aging and phase instability under the expected potential repository thermal conditions (CRWMS M&O 2000 [144229], Section 6.7). Data for Alloy-22 welds are not available yet. While additional data and analyses are being developed to better quantify the effects, it is assumed that the waste package outer barrier closure lid welds are subject to long-term aging and phase instability under the repository thermal conditions, and that their effects on the outer barrier corrosion can be represented with a corrosion enhancement factor. The enhancement factor was developed from the comparison of the passive current density data for non-aged Alloy-22 base metal samples to those of aged samples (aged at 700°C) (CRWMS M&O 2000 [144229], Section 6.7). The enhancement factor is represented with uniform distribution between 1 and 2.5.

Radiolysis-Induced Corrosion–The dominant contributor to dose rate at the waste package surface is from gamma radiation. Anodic shifts in the open circuit potential of stainless steel in gamma irradiated aqueous environments have been experimentally observed. The shift in corrosion potential was shown and subsequently confirmed to be due to the formation of hydrogen peroxide (CRWMS M&O 2000 [144229], Section 6.4.4).

Hydrogen peroxide additions (up to 72 ppm) to repository-relevant solutions in contact with Alloy-22 samples showed that the corrosion potential could shift a maximum of 200 mV in the anodic direction. This anodic shift is well below that required to cause breakdown of the passive film and initiation of localized corrosion (CRWMS M&O 2000 [151624], Section 3.1.6.6). Since extremely high radiation levels would be required to achieve such high hydrogen peroxide concentrations (CRWMS M&O 2000 [144229], Section 6.4.4), and the shifts in potential observed are less than those required for breakdown of the passive film, radiolysis will not initiate localized corrosion of the Alloy-22 waste package outer barrier (i.e., only general corrosion will occur). Furthermore, it has been shown that the rate of general corrosion would not be significantly affected by a shift of 200 mV in corrosion potential (CRWMS M&O 2000 [144229], Section 6.4.2).

Although the shift in corrosion potential for the Titanium Grade 7 drip shield material due to gamma irradiation (hydrogen peroxide) would likely differ from that of Alloy-22, the magnitude of the shift in potential due to gamma radiolysis would not be greater than that required to cause breakdown of the passive film and initiation of localized corrosion (CRWMS M&O 2000 [144971], Section 6.8). Similar to Alloy-22, gamma radiolysis will have no significant effect on general corrosion rates (CRWMS M&O 2000 [144971], Section 6.8) of titanium-based alloys. Therefore, the effects of radiolysis are excluded due to low consequence because gamma radiolysis does not initiate localized corrosion or have any significant effect on the rate of general corrosion and therefore, no significant effect on dose rate.

Hydrogen Induced Cracking–Atomic hydrogen generated at the surface of a metal can migrate into the metal and form metal hydrides causing the metal to be more brittle. The presence of the metal hydrides causes the metal to be more susceptible to cracking and thus to localized corrosion. Although hydrogen induced cracking is known to affect titanium, results from bounding analyses have indicated that hydrogen concentration in the drip shield will never surpass the threshold value for the onset of hydrogen induced cracking (CRWMS M&O 2000

[147640]). Thus, the drip shield will not be subject to hydrogen induced cracking. Hydrogen induced cracking of the waste container outer barrier (Alloy-22) is not considered to be a possible degradation mechanism under potential repository-relevant exposure conditions. Handbook data (ASM International 1987 [103753], pp. 650 to 651) indicate that fully annealed nickel-base alloys such as Alloy-22 may be immune to hydrogen-induced embrittlement (hydride cracking) (CRWMS M&O 2000 [148499]). The susceptibility to hydride cracking may be enhanced only when the strength level of this alloy is increased either by cold working or by aging at a temperature of 540°C at which ordering and/or grain-boundary segregation can occur. The susceptibility to cracking will be reduced with decreasing strength level and correspondingly with increasing aging temperature. However, since the waste package temperature will be sufficiently less than 540°C, the possibility of hydrogen induced cracking in Alloy-22 will be very remote (CRWMS M&O 2000 [136383]). Therefore, the waste package outer barrier will not be subject to hydrogen induced cracking.

3.4.2 Implementation of the Model

The computer implementation of the conceptual model provides a mechanism for incorporating the effects of the individual corrosion models in a probabilistic framework that captures the variability and uncertainty in the model parameters.

In the implementation, the effect of spatial and temporal variation in exposure conditions is incorporated by simulating corrosion processes for 400 drip shield and waste package pairs, each of which is assigned an exposure history consisting of an relative humidity and temperature profile. Profiles are selected from a suite of histories, which, as a whole, represent the range of relative humidity and temperature predicted to be found in the potential repository. The number of packages simulated (400) is considerably less than the projected final contents of the potential repository (11,770 packages [CRWMS M&O 2000 [136383], Attachment I]). This reduction was necessary for computational efficiency, but adequately represents the spatial variability over the entire repository as discussed in the WAPDEG Analysis Model Report (CRWMS M&O 2000 [146427], Sections 5.1 and 6.3.4).

Variability and uncertainty within each waste package and drip shield pair is incorporated by discretizing each component (drip shield and waste package) into numerous subareas called patches. Corrosion process parameters are sampled from probability distributions on a per-patch basis. Corrosion processes are then simulated as a function of time for each patch over a simulation period of 100,000 years.

General corrosion (both humid and aqueous) will only occur if the predicted relative humidity in the drift exceeds the critical relative humidity for the material. A table of critical relative humidity values as a function of temperature is used in this determination (CRWMS M&O 2000 [146427], Section 6.3.8). At each simulated time, predicted relative humidity and temperature values are extracted from the RH and thermal history specified for each drip shield and waste package pair. If the predicted relative humidity value exceeds the critical RH at the predicted temperature, then general corrosion proceeds.

Results of the analyses consist of patch and package failure histories as a functions of time. These results provide input to the waste form degradation models and EBS transport models to determine releases from the waste package to the environment.

3.4.2.1 WAPDEG

The TSPA-SR subsystem model for evaluating degradation of the waste package and drip shield is the WAste Package DEGradation (WAPDEG) model (CRWMS M&O 2000 [146427]). WAPDEG is based on a stochastic simulation approach and provides a description of waste package and drip shield degradation, which occurs as a function of time and potential repository location for specific design and thermo-chemical-hydrologic exposure conditions. Waste package and drip shield degradation are, for the most part, independent. Exceptions include the transition from humid air to aqueous general corrosion on the waste package exterior after drip shield failure, and the calculation of fluid inflow to a failed waste package is dependent upon the degree of failure of the associated drip shield.

In the current analysis, the waste package degradation model is composed of two components; the WAPDEG dynamic-link library (WAPDEG.DLL) which models the variability in waste package degradation, and the GoldSim implementation which models the uncertainty in the parameters passed to the WAPDEG DLL. Further details of the WAPDEG-GoldSim interface can be found in the TSPA-SR Model Document (CRWMS M&O 2000 [148384], Section 6.3.3).

3.4.2.2 Gaussian Variance Partitioning

General corrosion rate data for the drip shield and waste package are described by probability distributions that reflect both uncertainty and variability in the parameter. "Uncertainty" describes the lack of knowledge concerning the exact corrosion initiation threshold and degradation rate of a material, while "variability" refers to the differing corrosion initiation thresholds and different degradation rates that could occur because of different or varying material properties (on a microstructural scale) and temporal or spatial exposure conditions.

The uncertainty and variability components of the general corrosion rate distributions are separated through the use of Gaussian Variance Partitioning (GVP) (see CRWMS M&O 2000 [151624], Section 3.2.2 for a more thorough discussion). WAPDEG then samples from the resulting variability distribution the corrosion rate parameter values for each waste package patch. In summary, GVP separates the input general corrosion rate cumulative distribution function (CDF), containing both uncertainty and variability, into two separate distributions, one that characterizes variability and another that characterizes uncertainty. Each distribution has only a fraction of the input CDFs total variance (i.e., if the fraction of variance due to uncertainty is U, then the fraction due to uncertainty is 1-U). The median value of the variability distribution is sampled from the uncertainty distribution. The percentage of the total variance due to uncertainty is itself uncertain and is sampled from a uniform distribution between 0 and 1. The quantile at which to sample the median general corrosion rate is also uncertain and is sampled from a uniform distribution between 0 and 1.

Figure 3.4-13 shows, along with the input general corrosion CDF, the resulting variability CDFs for the Alloy-22 waste package outer barrier using 25 percent to 75 percent, 50 percent to

50 percent and 75 percent to 25 percent uncertainty and variability partitioning ratios and the 50th uncertainty quantile for the median of the variability distributions. Figure 3.4-14 shows the same general corrosion rate CDFs for the Titanium Grade 7 drip shield. Figures 3.4-15 and 3.4-16 show the variability CDFs for the Alloy-22 waste package outer barrier using 25 percent, 50 percent and 75 percent variability and the 25th and 75th uncertainty quantiles, respectively, for the median of the variability distributions.

3.4.2.3 Drip Shield Implementation

For modeling simplicity, the variability in drip shield degradation is assumed to be adequately characterized by modeling 500 patches for a 15 mm thick drip shield. Figure 3.4-17 illustrates the discretization used. The validity of this assumption was ascertained through sensitivity testing where the number of patches was varied using values of 200, 500, and 1,000 patches per drip shield. Results for the 500 patch and 1,000 patch case were found to be sufficiently similar to justify use of the 500 patch value on the basis of increased computational efficiency (CRWMS M&O 2000 [146427], Section 6.3.3).

The general corrosion rate distribution for Titanium Grade 7 (based on measured general corrosion rates for the analog Titanium Grade 16, see Section 3.4.1.2) is partitioned using Gaussian Variance Partitioning as described in the previous section. Gaussian Variance Partitioning is applied twice, once to provide probability distributions for dripping conditions (i.e., aqueous corrosion) applied to the drip shield top, and once to provide distributions for humid-air corrosion applied to the underside of the drip shield. The appropriate distributions are then sampled to provide corrosion rates for outside-in (from the dripped-on exterior of the drip shield to the shielded interior) and inside-out (from the interior to the exterior) corrosion for each patch. Throughout the simulation, the sampled dripping condition general corrosion rate is applied to the drip shield exterior surface and the sampled humid-air general corrosion rate is applied to the drip shield interior.

3.4.2.4 Waste Package Implementation

The waste package is modeled as a cylinder with a total of 1,000 patches as illustrated schematically in Figure 3.4-18. Based on the sensitivity assessment conducted for the drip shield (CRWMS M&O 2000 [146427]), it was concluded that using twice as many patches would adequately represent the spatial variability in waste package degradation. The use of 1,000 waste package patches is conservative relative to the 938 waste package patches used in the analyses documented in the WAPDEG AMR (CRWMS M&O 2000 [146427]) as a larger number of patches results in a larger number of samples from each distribution used in modeling waste package degradation.

3.4.2.5 Waste Package - General Corrosion

The design of the waste package requires a more complex approach to modeling general corrosion than that used for the conceptually simpler drip shield. To allow a consistent approach over the package body and the closure lids, the waste package outer wall was divided into two conceptual layers: an inner pseudo-layer, and an outer pseudo-layer, providing an equivalent to the outer- and middle closure lids (CRWMS M&O 2000 [146427]).

General corrosion can be outside-in (from the exterior of the package towards the interior) and, after the first complete patch failure allows liquid into the interior of the package, inside-out (from the package interior towards the exterior). The general corrosion rate distribution for Alloy-22 is partitioned using Gaussian Variance Partitioning. The appropriate distributions are then sampled to calculate corrosion rates for each direction for each patch.

3.4.2.6 Waste Package - Stress Corrosion Cracking

Patches at the edges of the closure lid corresponding to closure lid welds incorporate stress corrosion cracking through the slip dissolution model (CRWMS M&O 2000 [148375]). It was recommended that patches subject to stress corrosion cracking contain a single initial crack (CRWMS M&O 2000 [151624]) to capture details of individual crack growth. However, because WAPDEG requires that all patches be the same size, this would have resulted in approximately ten thousand patches per waste package, which in turn would have resulted in unacceptably poor computational performance. Consequently, the current patch size was selected, and ten stress corrosion cracking cracks were simulated for each closure lid weld patch. This is a conservative assumption from one perspective, as all ten cracks are considered failed when the first of the ten cracks penetrate, leading to a greater failed area than if a single crack had been used.

Each patch subject to stress corrosion cracking is initialized with ten incipient cracks of 50 μm depth. Crack growth parameters are sampled from appropriate distributions. SCC is initiated after general corrosion starts on the patch and if the predicted stress at an incipient crack location is greater than the threshold stress. It continues to grow as long as the predicted stress intensity at the crack tip exceeds the threshold stress.

Stress and stress intensity factor profiles are calculated once per realization and are applied to all waste packages. A profile is provided for each patch and varies according to the patch position around the circumference of the package.

Initiation of SCC requires general corrosion. Consequently, middle lid SCC can not start until at least one patch has failed on the outer lid (either through general corrosion or SCC), allowing humid air or drips to penetrate the air gap between the outer and middle lids.

3.4.2.7 Waste Package - Manufacturing Defects

The manufacturing defect abstraction model (CRWMS M&O 2000 [146427], Sections 3.2.5, 4.1.7, 6.3.11) is run before the start of the time dependent corrosion models. The model is executed twice for each package; once for the middle lid, and once for the outer lid. Parameter values are sampled from appropriate distributions (CRWMS M&O 2000 [144551]). Predicted surface-breaking undetected flaws are then allocated to closure-lid weld patches as initial cracks. These cracks are generally of much greater depth than the incipient cracks initialized for stress corrosion cracking. Defect crack growth is then simulated with the slip dissolution model (refer back to Section 3.4.1.4 for additional discussion).

3.4.2.8 Waste Package - Other Modes

Microbially induced corrosion is described by a general corrosion rate enhancement factor characterized by a uniform distribution ranging from 1 to 2 (CRWMS M&O 2000 [144229]). Similarly, the general corrosion rate enhancement factor for thermal aging and phase instability is described by a uniform distribution ranging from 1 to 2.5 (CRWMS M&O 2000 [144229]).

Once the critical relative humidity threshold for a waste package is exceeded, the distributions for each mode are sampled and the general corrosion rate over all patches enhanced by the sampled multipliers.

3.4.3 Results and Interpretation: Evaluation of Key Issues and Importance to Performance

All input files used in this analysis and output files produced from this analysis are tracked by DTN: MO0007MWDTSP01.003 [151706]. The performance of the waste package and drip shield was simulated by executing 300 realizations of the nominal case (or base case) scenario. Each realization corresponds to a complete simulation of 400 drip shield and waste package pairs (see Section 3.4.2) over the time interval from potential repository closure to 100,000 years subsequent to closure. In the nominal case presented here, the conservative conceptual model of SCC uncertainty is used (i.e., the uncertainty fraction of yield strength is set to 30 percent). In Section 5.2.3.2, simulation results utilizing a range of stress corrosion cracking uncertainty model parameters are presented. The analysis results are presented for the upper and lower bounds, median, 95th, 75th, 25th and 5th percentiles and mean as a function of time for the following output parameters:

- Drip shield first breach (or failure) Figure 3.4-19 (top plot)
- Waste package first breach (or failure) Figure 3.4-19 (middle plot)
- Waste package first patch penetration Figure 3.4-19 (bottom plot)
- Drip shield patch penetration percentage per failed waste package Figure 3.4-20 (top plot)
- Waste package patch penetration percentage per failed drip shield Figure 3.4-20 (bottom plot).

The first breach or failure time is defined, for the purposes of this section, as the earliest time at which either a pit, crack, or patch has penetrated the barrier under consideration (the Titanium Grade 7 drip shield or Alloy-22 waste package outer barrier). Note that localized corrosion does not initiate for either the waste package (Alloy-22 outer barrier) or the drip shield, because the exposure conditions on the drip shield and waste package surface are not severe enough to initiate localized corrosion. Also note that the drip shield is assumed not to be subject to SCC, therefore no crack penetration failure of the drip shield is presented. For the drip shield, the first patch breach time profile is the same as the failure time profile.

Most of the following discussion focuses on the upper bound and median failure curves. The upper bound curves describe very low probability results, while the median is the most likely case with fifty percent of the realizations showing earlier failure times. Low probability results are, by definition, probably not representative of the actual performance of the drip shield and waste package system, but do provide a useful bounding case. In other words, if the low-probability results are acceptable, the actual performance is more than adequate.

The upper-bound drip shield failure curve on Figure 3.4-19 (top plot) indicates that the first drip shield patch fails at about 20,000 years. Additionally, the upper bound case indicates that half of the drip shields have failed within 1,000 years after the initial failure, and all have failed by 30,000 years. The upper-bound drip shield failure curve on Figure 3.4-20 (top plot) indicates that all patches on all failed drip shields have failed by 100,000 years. As all drip shields have failed by 30,000 years, this means that the upper-bound curve predicts that all drip shields have degraded in their entirety by 100,000 years.

Initial failure time for the median case is similar, however the failures occur over a longer duration.

Figure 3.4-19 (middle plot) shows that the first waste package failure on the upper-bound curve occurs at approximately 10,000 years. Comparing the overall waste package failure to the waste package patch failure upper bound curves indicates that this first failure is due to a middle-lid crack failure. This initial failure is likely a result of one or more manufacturing defects in a single package. The upper-bound waste package patch failure curve (Figure 3.4-19 bottom plot) shows that it takes approximately 100,000 years for at least one patch failure to occur in all waste packages. Figure 3.4-20 (bottom plot) shows that approximately 2 percent of the waste package patches have failed at this time.

Results for the median case show an initial failure at approximately 40,000 years and about 0.12 percent of the patches have failed by 100,000 years.

The nominal case results indicate that the drip shield and waste package system performs exceptionally well, effectively isolating the waste from the natural environment for tens of thousands of years.

3.5 WASTE FORM DEGRADATION

The waste form degradation model evaluates the interrelationship among the in-package water chemistry, the degradation of the waste form (including cladding), and the mobilization of radionuclides. Specifically, the waste-form degradation model consists of the following components (Figure 3.5-1) that

- Define the radioisotope inventories for the CSNF and codisposal waste packages-Inventory Abstraction
- Evaluate in-package water chemistry—In-Package Chemistry Component Abstraction
- Evaluate the matrix degradation rates for CSNF, DSNF, and HLW waste forms—Waste Form Matrix Degradation Component Abstractions

- Evaluate the rate of Zircaloy cladding degradation (in the case of the CSNF)—Cladding Degradation Component Abstraction

- Evaluate the radionuclide concentrations for aqueous phases—Dissolved Radionuclide Concentration Component Abstraction

- Evaluate the waste form colloidal phases—Colloidal Radionuclide Concentration Component Abstraction.

The model developed for the TSPA-SR is applicable to three generic waste form categories: (1) CSNF, (2) DSNF, and (3) HLW glass. As described in Section 3.4, these three categories are contained and disposed of in two types of waste packages—CSNF waste packages and codisposal waste package, with the latter one containing both DSNF and HLW glass.

For both the CSNF and co-disposal waste packages, the waste form degradation model describes the evolution of the chemical environment, corrosion of the protective cladding leading to perforations and cladding failure by unzipping ("splitting") in the case of CSNF, dissolution of the exposed fuel matrix, and finally mobilization of the radionuclides. These processes are shown schematically for CSNF in Figure 3.5-2. The calculated radionuclide release rates from waste form are, in turn, provided to the EBS transport model (Section 3.6), which calculates the radionuclide releases from the EBS.

The waste form degradation model is primarily designed for the nominal or reference scenario (defined in Section 1.6) but is also used as a source term for the igneous disruptive scenario and human intrusion scenario. The subcomponents are computationally linked in a sequential manner and, therefore, treated as uncoupled. There is one instance, however, of a weak feedback mechanism between certain subcomponent models. Specifically, the in-package chemistry is dependent upon the amount of CSNF exposed and the alteration rate of the HLW borosilicate glass. This coupling is accounted for in the TSPA-SR calculations by lagging the feedback by one time step[1] (i.e., the waste form degradation model does not iterate during the time step).

The conceptual models developed for waste form degradation are described in the *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332]) which summarizes the results of theoretical and experimental studies, described in numerous analysis/model reports (AMR) on the degradation of the three general waste forms: CSNF, DSNF, and HLW. A brief summary of the models, similar to that provided in the waste form degradation *PMR* is also provided here to help in understanding the results of the TSPA-SR. An important purpose of the *Waste Form Degradation PMR* and underlying AMRs is to provide the basis of the models and explain the appropriateness of models for their intended use in the TSPA-SR. This validation information is not provided herein. Additional information on the implementation and validation of these conceptual model components in the TSPA-SR computer model can be found in the *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]), and *Total System Performance Assessment-Site Recommendation Methods and Assumptions* (CRWMS M&O 1999 [105017]).

[1] The size of the time step depended upon the length of the simulations. For all simulations up to 100,000 years, the time step was 500 years. For simulations run to 1,000,000 years, the time step at 100,000 years increased to 1,000 years, then 2,000 years, and finally 4,000 years.

Figure 3.5-3, a schematic summary of the waste form degradation model, identifies the major inputs and outputs, key subcomponents of the model, and the experimental bases for confidence in the model. The waste form degradation model receives time varying conditions for the waste package surface temperature and water volume entering the waste package (Section 3.3). The primary outputs of the waste form degradation model consist of the dissolved and colloidal concentrations of radionuclides, which are provided to the EBS transport model (Section 3.6).

3.5.1 Inventory Abstraction

The model abstraction for the waste inventory defines the source term for the CSNF and codisposal waste packages in terms of both the quantity and spectrum of radioisotopes. This information is used with the abstraction for waste form degradation to determine the mobilization of the radionuclides. Radioisotopes contained in the waste packages include fission products from reactor operations, actinides from neutron capture in uranium and plutonium, and activation products from neutron irradiation of structural materials and trace elements. Altogether, these fission products, actinides, and activation products constitute well over 100 radioisotopes that may be collectively present in the waste packages at the time of the potential repository closure.

Many of the radioisotopes, however, have intrinsic physical and chemical properties (e.g., very short half-life, low solubility, or strongly sorbing characteristics), and/or small inventory that prevent them from posing a radiological risk to a receptor group at the point of compliance as discussed in Section 1.3. As a result, only a small set of radionuclides needs to be considered in the evaluation of postclosure performance. To develop the abstraction for the radioisotope inventory, the following tasks were conducted:

- Grouping the various spent fuel types and waste forms into three generic categories of CSNF, DSNF, and HLW and the two waste package types of CSNF and codisposal waste

- Evaluating an average radioisotope inventory for each waste form category and waste package type

- Selecting the radioisotopes that are most important for calculation of the expected annual dose.

With respect to the mandated disposal capacity of 70,000 MTHM, the CSNF waste form would contribute about 63,000 MTHM (about 90 percent of the total waste), the total DSNF waste would be 2,333 MTHM (or 3.33 percent), and the HLW glass waste would be about 4,667 MTHM (or 6.67 percent). The technical basis for the inventory model is documented in the *Draft of AMR Inventory Abstraction* (CRWMS M&O 2000 [152218]).

3.5.1.1 Conceptual Model

The radioisotope inventory component provides an estimate for radioisotope activities in containers destined for disposal in the potential Yucca Mountain repository based on the radionuclide activity for the grouping of various fuel types for CSNF, DSNF, and HLW shown in Figure 3.5-4. Because the amount of plutonium disposition waste is so small (relative to the other waste forms), its inventory was included in the CSNF and HLW, as indicated in Figure 3.5-4. As currently modeled, there are 7,860 CSNF waste packages and 3,910 codisposal waste packages placed in the potential geologic repository (CRWMS M&O 2000 [152218]).

Basis for Commercial Spent Nuclear Fuel Inventory Projection–Commercial nuclear power plants use a variety of fuels and fuel configurations in their reactor cores to generate power. In the United States, the current nuclear fuel is enriched uranium dioxide (UO_2), but a plutonium and uranium mix may also be used in the future. Fuel pellets are packed into long cylindrical fuel rods (varying in size, depending on the design), and these fuel rods (clad in Zircaloy or stainless steel) are then bundled into assemblies. The number of fuel rods per assembly and the number of assemblies in a reactor core vary, depending on the core and reactor design (i.e., PWRs or BWRs).

As currently projected, about 230,000 CSNF assemblies will need to be disposed in the potential geologic repository (CRWMS M&O 2000 [152218]). Each assembly, depending on the reactor configuration, initial fuel enrichment, burn-up, and the age of the waste (time in storage), will have a unique isotopic composition. Average isotopic compositions were developed using historical data supplied by utility companies. The most up to date historical data consists of reactor reported assembly discharges through December 1995 from their reactors. The utility companies also provided a forecast for the assembly discharges over the next 5 reloading cycles, which occur about every 1.5 years. From this information, DOE developed alternative schedules for shipping the assemblies to the potential Yucca Mountain repository, one of which was selected for estimating waste streams.

Radionuclide inventories for each assembly in a waste stream were then estimated, and the 230,000 CSNF assemblies were grouped into five proposed waste package configurations (The two largest groups out of the five were 4500 packages containing 21 assemblies of PWR fuel and 3000 packages containing 44 assemblies of BWR fuel). An overall average radionuclide inventory for all the CSNF waste packages was then computed by weighting by the number of packages in each of the five waste package configurations (CRWMS M&O 2000 [152218]).

Basis for U.S. Department of Energy-Owned Spent Nuclear Fuel and High-Level Radioactive Waste Inventory Projection–In the current reference potential repository design, the DSNF and HLW forms will be disposed together in codisposal waste packages. Therefore, they are discussed together here. The DSNF waste form category encompasses more than 250 distinct types of spent fuel, with radionuclide inventories that vary widely, depending on the reactor history of the fuel (DOE 1999 [107790]). The DSNF waste form will be packaged in four types of canisters (consisting of two lengths and two diameters) before they are shipped to the potential Yucca Mountain repository for disposal. An average radionuclide inventory for the DSNF waste form was then calculated by using the estimated inventories for each of five codisposal configurations (one configuration using long DSNF canisters and long HLW canisters, one using short DSNF and short HLW canisters, one using short DSNF canisters and long HLW canisters, and one with only HLW canisters).

The naval SNF will be placed in two types of canisters (consisting of two lengths) and placed in a waste package without any HLW. Additional analysis of Naval SNF that has been conducted is summarized in Section 3.5.3.

The proposed technology for immobilization of reprocessed defense waste is vitrification in a borosilicate glass. Significant quantities of glass waste are currently produced and stored at the Savannah River Site. Production of HLW glass is also anticipated to start at the Hanford Reservation and the Idaho National Engineering and Environmental Laboratory. Finally, a small amount of HLW glass was produced at West Valley in New York, but the exact amount to be disposed has not been finalized. These generator sites provided radionuclide inventories for the HLW representative of their vitrification process. This information was used to calculate an average radionuclide inventory for the short and long HLW canisters.

Radionuclide Screening and Selection–The combined list of radionuclide for the three waste allocation categories was screened to identify the specific ones that could potentially make significant contributions to the calculation of expected annual dose. A radionuclide screening procedure (CRWMS M&O 2000 [152218]) was used that considered the following factors: relative contribution to annual dose, radionuclide longevity (i.e., decay and production), elemental solubility, transport affinity, release scenario, and containment time (e.g., 10,000 years and 1,000,000 years). This screening procedure produced an initial list of radionuclides that were then augmented to account for ingrowth of the actinide decay chains.

The results of the screening identified important radionuclides for various scenarios and time periods:

- **Nominal Scenario (and Indirect Release for Volcanism Scenario), 10,000 Years**– ^{227}Ac, ^{241}Am, ^{243}Am, ^{14}C, ^{129}I, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{99}Tc, ^{229}Th, ^{232}U, ^{233}U, ^{234}U, ^{236}U, and ^{238}U

- **Nominal Scenario (and Indirect Release for Volcanism Scenario), 1,000,000 Years**– The nominal scenario set for 10,000 years, plus ^{231}Pa, ^{210}Pb, ^{242}Pu, ^{226}Ra, and ^{230}Th

- **Volcanism Scenario with Direct Release, 10,000 Years**–^{227}Ac, ^{241}Am, ^{243}Am, ^{137}Cs, ^{231}Pa, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{90}Sr, ^{229}Th, ^{232}U, ^{233}U

- **Volcanism Scenario with Direct Release, 1 Million Years**–Volcanism scenario with direct release for 10,000 years, plus ^{237}Np[2], ^{210}Pb, ^{242}Pu, ^{226}Ra, and ^{230}Th

[2] Plutonium dominates the expected annual dose from direct releases up to 100,000 years. Thereafter ^{237}Np can become important. However, the TSPA-SR simulation of the igneous disruption scenario only considers a 100,000 year period, since after ~2,000 years, the expected annual dose from indirect releases (e.g., groundwater flow through disrupted waste containers) is greater than the direct releases. As noted above ^{237}Np is considered in the nominal scenario and in the igneous intrusion scenario for indirect releases.

- **Human Intrusion Scenario, 10,000 Years**–^{227}Ac, ^{241}Am, ^{243}Am, ^{14}C, ^{137}Cs, ^{129}I, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{90}Sr, ^{99}Tc, ^{229}Th, ^{232}U, ^{233}U, ^{234}U, ^{236}U, and ^{238}U

- **Human Intrusion Scenario, 1 Million Years**–Human intrusion scenario set for 10,000 years, plus ^{231}Pa, ^{210}Pb, ^{242}Pu, ^{226}Ra, and ^{230}Th.

With regard to the human intrusion scenario and the 10,000 year compliance period, ^{63}Ni was initially identified in the *Draft of AMR Inventory Abstraction* (CRWMS M&O 2000 [152218]) as a potentially important radionuclide. Subsequent evaluations, however, have shown that ^{63}Ni can be screened out because of its short half-life (100.1 years) relative to the nuclide residence time and its sorption in the SZ. For time periods of a 1 million years, ^{235}U was subsequently added to the list because it is a source for ^{227}Ac, which was considered potentially important to dose. In addition, certain radioisotopes were added because of their relevance to the groundwater protection standard (proposed 40 CFR Part 197 [64 FR 46976 [105065]]). This standard specifies concentration limits for ^{226}Ra and ^{228}Ra. Consequently, ^{228}Ra and its precursor, ^{232}Th, were added to the list for the case of the nominal scenario. These adjustments expanded the list to a total of 26 radioisotopes for the TSPA-SR. Dose conversion factors were calculated for a selection of the 26 radioisotopes listed as described in Section 3.9.2 (Table 3.9-2) and Section 3.10.3 (Table 3.10-8). Obviously, radionuclides that were necessary only for evaluating chains (e.g., ^{235}U) did not require dose conversion factors. Furthermore, radionuclides only important for the simulations out to 1 million years and only contributed a small portion to the dose used dose conversion factors that had been used for previous TSPAs.

3.5.1.2 Implementation in the Total System Performance Assessment

The computer implementation of the inventory abstraction is a simple table look-up of the quantity of radionuclides at the time of waste emplacement for the CSNF and co-disposal waste packages. The inventories for each of 26 radionuclides are listed in Table 3.5-1 for both the CSNF and codisposal waste packages. These inventories are adjusted for decay, in-growth, and release from the waste form during the TSPA-SR simulations.

While most of the radioisotopes in the CSNF waste are bound in the UO_2 matrix, some fission product gases such as ^{137}Cs, and ^{129}I are known to migrate to the gap between the matrix and cladding. Furthermore, many radionuclides can migrate from the matrix to the cooler grain boundary. This migration process is important, because these radionuclides are released much faster than those bound in the fuel matrix. To account for the distinct release rates, the inventory for the CSNF waste form is divided into two parts: (1) matrix and (2) fast release. The fast release inventory is the sum of the gap inventory and grain boundary inventory. The grain boundary inventory is assumed to be the total inventory of each radionuclide times a fraction sampled from a uniform distribution between 0 and 0.004. As explained further in the *Waste Form Degradation PMR* and supporting AMR on cladding degradation, the distribution is based on the fractions of radionuclides observed to be released in short term tests (less than 200 days) and extrapolating to 5 years (CRWMS M&O 2000 [138332]; CRWMS M&O 2000 [147210]. The gap inventory is an additional 0.042 and $1/3 \times 0.042$ of the inventory of ^{137}Cs and ^{129}I, respectively.

Table 3.5-1. Average Inventories of Commercial Spent Nuclear Fuel and of Co-disposal Waste Packages

Radioisotope	CSNF Mass per Package (g/pkg)	Co-disposal Mass per Package (g/pkg)	
		DSNF	HLW
$^{227}Ac^{a}$	3.09×10^{-6}	1.13×10^{-4}	4.67×10^{-4}
^{241}Am	1.09×10^{4}	1.17×10^{2}	6.57×10^{1}
^{243}Am	1.29×10^{3}	1.49	3.99×10^{1}
^{14}C	1.37	4.96×10^{-2}	6.43×10^{-3}
$^{137}Cs^{b}$	5.34×10^{3}	1.12×10^{2}	4.51×10^{2}
^{129}I	1.80×10^{3}	2.51×10^{1}	4.80×10^{1}
^{237}Np	4.74×10^{3}	4.79×10^{1}	7.23×10^{1}
$^{231}Pa^{c}$	9.87×10^{-3}	3.25×10^{-1}	7.96×10^{-1}
$^{210}Pb^{a,c}$	0.00	1.40×10^{-8}	1.14×10^{-7}
^{238}Pu	1.51×10^{3}	6.33	9.33×10^{1}
^{239}Pu	4.38×10^{4}	2.30×10^{3}	3.89×10^{3}
^{240}Pu	2.09×10^{4}	4.89×10^{2}	3.81×10^{2}
$^{242}Pu^{c}$	5.41×10^{3}	1.11×10^{1}	7.77
$^{226}Ra^{a,c}$	0.00	1.87×10^{-6}	1.67×10^{-5}
$^{228}Ra^{a}$	0.00	6.98×10^{-6}	3.19×10^{-6}
$^{90}Sr^{b}$	2.24×10^{3}	5.54×10^{1}	2.88×10^{2}
^{99}Tc	7.68×10^{3}	1.15×10^{2}	7.29×10^{2}
^{229}Th	0.00	2.66×10^{-2}	4.08×10^{-3}
$^{230}Th^{c}$	1.84×10^{-1}	1.06×10^{-2}	7.82×10^{-3}
^{232}Th	0.00	1.49×10^{4}	7.31×10^{3}
$^{232}U^{b}$	1.01×10^{-2}	1.47×10^{-1}	8.23×10^{-4}
^{233}U	7.00×10^{-2}	2.14×10^{2}	1.11×10^{1}
^{234}U	1.83×10^{3}	5.72×10^{1}	4.72×10^{1}
^{235}U	6.28×10^{4}	8.31×10^{3}	1.70×10^{3}
^{236}U	3.92×10^{4}	8.53×10^{2}	3.98×10^{1}
^{238}U	7.92×10^{6}	5.09×10^{5}	2.61×10^{5}

Source: CRWMS M&O 2000 [152218]

NOTES: [a]These radionuclides are not transported in nominal groundwater scenario; rather, the inventory in the biosphere is determined by assuming secular equilibrium

[b]These radionuclides are not transported in nominal groundwater scenario.

[c]These radionuclides are included for TSPA-SR calculations for time periods beyond 10,000 years.

3.5.1.3 Treatment of Uncertainty and Variability

The current inventory analysis for TSPA-SR is more detailed and flexible than done for previous TSPAs. The principle source of uncertainty and variability in the current inventory analysis is in the projections of waste form inventories. The radionuclide inventories were derived from estimates of future waste streams. The actual waste streams may differ with respect to fuel burn-ups, fuel ages, fuel enrichments, and reactor efficiencies. However, changes that might occur are expected to change the projected inventories by only small amounts (DOE 1999 [107790]; CRWMS M&O 1999 [119348]). The variation is small in comparison to the uncertainties and variabilities in other subcomponents of the waste form model. Of the 230,000 commercial assemblies modeled in the TSPA-SR, the curies of important radionuclides varies only an order of magnitude. The range in the codisposal packages varies more (the range in the HLW canisters varies two orders of magnitude and in the DSNF canisters by five orders of magnitude), but the total inventory is so much smaller in the codisposal packages (as mentioned in the next section and Section 5.2). The variation of curies in the CSNF packages is much less than the variation in other components (e.g., see Section 3.5.4.4 on CSNF cladding unzipping rates which has a three order of magnitude variation). Therefore, the model abstraction for radionuclide inventory is deterministic and does not include statistical distributions to express data uncertainty or variability. With regard to the selection of important radionuclides, the radionuclide screening procedure is considered sufficiently conservative in that it identifies a larger set of radionuclides than would actually be needed in order to appropriately determine the expected annual dose.

3.5.1.4 Results and Interpretation

The model abstraction for the inventory in the CNSF and co-disposal waste packages prescribes the initial activity of each radionuclide and describes the variation of the activity with time. For certain radionuclides, the activity decreases with time as a result of simple radioactive decay, while for other radioisotopes the activity increases with time because of ingrowth. For those radionuclides in the fission product category, the model abstraction calculates the variation of activity with time in accordance with the first order decay law. For the radionuclides that fall into the category of actinide elements, the model abstraction accounts for the decay sequence and calculates the activity of each radioisotope as a function of decay and ingrowth.

The inventory of short-lived fission products such as ^{90}Sr and ^{137}Cs, for example, decreases substantially over the 10,000-year compliance period. In contrast, the inventories of long-lived fission products such as ^{14}C, ^{99}Tc, and ^{129}I decrease only slightly over the compliance period. With regards to actinide elements, many of the radionuclides in Table 3.5-1 are components of simplified representations of actinide element decay chains (i.e., actinium, thorium, neptunium, and uranium series). The simplified decay chains used in the model abstraction are illustrated in Figure 3.5-5 and are mathematically described by the Bateman equations.

For the case where the nuclear waste remains contained in the waste packages, the decay history of each radioisotope can be analytically calculated in accordance with their decay sequence and half-lives. The graphs in Figure 3.5-6 show the decay histories for selected sets of radionuclides over a period of 1,000,000 years. In these graphs, the initial activity of each radionuclide is the activity disposed in the CNSF and co-disposal waste packages divided by the mandated disposal limit of 70,000 MTHM.

As easily calculated from Table 3.5-1 and the total number of waste packages of each type, the total inventory of ^{237}Np and ^{99}Tc in the co-disposal is more than an order of magnitude less than in the CSNF packages. Furthermore, DSNF only contributes only 2 percent of the ^{237}Np and ^{99}Tc inventory in the co-disposal packages. Hence, the CSNF packages potentially have much greater influence on the dose provided the release rates are similar. As discussed in

Section 3.5.3, the release rates are indeed similar such that CSNF provide roughly an order more to the total dose than co-disposal packages as discussed later in Section 5.2; hence, variability of the inventory in the co-disposal can be much greater and still not influence the total dose.

3.5.2 In-Package Chemistry Component Abstraction

The in-package chemistry component models the evolution of the water chemistry inside the failed waste package as a function of water inflow rate and waste package and waste form corrosion rate. The water chemistry characteristics are primarily pH, ionic strength, and total carbonate concentration (assuming that the partial pressure of oxygen and carbon dioxide are held constant at atmospheric conditions). Additional chemistry characteristics include the concentrations of fluoride and chloride. This water chemistry information is used by five other waste form degradation components, which are dependent on the in-package water chemistry. Specifically, the waste form matrix degradation rate for CSNF and HLW, the dissolved concentration of radioisotopes, stability of colloids, and degradation of CSNF cladding are dependent on water chemistry parameters. The rates of degradation of the waste matrix and the stainless steel within the waste package, in turn, influence the water chemistry, and so there is a coupling among all the chemically interacting components of the system. However, the coupling does not usually involve feedback in the TSPA-SR. The only feedback is between the in-package chemistry component, and the cladding and HLW degradation components as noted in the Introduction to Section 3.5.

The in-package chemistry model provides a quantitative description of the fluid chemistry for all three TSPA-SR scenarios: (1) nominal, (2) igneous disruptive event, and (3) human intrusion (e.g., refer to Table 4.4-3). The technical basis for the in-package chemistry model abstraction is documented in *In-Package Chemistry Abstraction* (CRWMS M&O 2000 [129287]).

3.5.2.1 Conceptual Model

The conceptual model for the in-package chemistry includes the processes of water flow into and out of the failed waste package, water interactions with the waste form and waste package materials, and the resulting dissolution or precipitation and complexation reactions. In the in-package chemistry component, the failed waste package is idealized as a simple mixing-cell. The fluid chemistry inside the package is dependent upon the initial chemical composition of the water entering the waste package, the flow rate through the package, and the volume of water in the waste package. In addition, the in-package chemistry component takes into account the degradation of the contents of the waste package. The contents important to evaluating the water chemistry include the borosilicate glass encapsulating HLW, the uranium dioxide fuel (UO_2), 316 stainless steel, and A516 low carbon steel.

The conceptual model is also constrained by the general assumptions and technical bases of the process model, EQ3/6 (CRWMS M&O 1998 [102837]), which was used to simulate the in-package chemical conditions. In applying the EQ3/6 (CRWMS M&O 1998 [102837]), assumptions made in the conceptual model for the process model include:

- Chemical composition of incoming water was analogous to that of J-13 water (Harrar et al. 1990 [100814]).

- Water flux into the failed waste package was constant during the process model simulation; three rates, based on the TSPA-VA[3] were used: 0.15, 0.015, and 0.0015 m^3/yr.

- All voids in the waste form were filled with water.

- Partial pressures of O_2 and CO_2 were set at $10^{-0.7}$ and 10^{-3} bars, respectively.

- Water temperature was fixed at 25°C.

- Cladding coverage was set at either 99 percent, 20 percent, or 1 percent.

- The DSNF represented by the Fast Flux Test Facility (FFTF) SNF.

In developing the conceptual model for in-package chemistry, two types of waste packages were modeled: the CSNF and co-disposal waste packages. As shown in Figure 3.5-7, the time of failure of container groups is considered in determining ranges of values for pH. Carbonate concentration and Eh is determined with input from pH values. In addition to pH, the principal outputs of the model are redox conditions, ionic strength, total carbonate concentration, and fluoride and chloride concentrations.

From the analysis of the EQ3/6 (CRWMS M&O 1998 [102837]), simulation results, the pH was found to be a very important parameter. Therefore, pH was included in developing abstractions for other chemical parameters, such as ionic strength, total carbonate concentration, and others. Because of its dependence on other physical and chemical factors that change with time, pH in the failed package also varied with time. In fact, the pH was found to vary rapidly during the first 1,000 years, reaching its minimum value during this time period. Immediately after 1,000 years, the pH calculations approach a near constant value. For simplicity, the pH analysis was divided into two time periods: the first 1,000 years (early) and the post 1,000 years (late).

3.5.2.2 Implementation in the Total System Performance Assessment

The chemical parameters produced with the EQ3/6 (CRWMS M&O 1998 [102837]; Wolery and Daveler 1992 [100097]) code were abstracted into simple regression equations. For example, the pH of the fluid solution was mathematically modeled by a regression equation with functional dependence on (1) water flux, (2) waste package corrosion rate, and (3) cladding coverage (for the CSNF waste package) or glass dissolution rate (for the co-disposal waste package). For CSNF packages, EQ3/6 (CRWMS M&O 1998 [102837]; Wolery and Daveler 1992 [100097]) calculations were performed for three water-flux rates, three cladding coverage fractions, and a representative and maximum corrosion rate of material within the waste packages (CRWMS M&O 2000 [129287]). The process model results of pH using EQ3/6 (CRWMS M&O 1998 [102837]; Wolery and Daveler 1992 [100097]) were modeled as planar surfaces in three-dimensional space (Equation 3.5-1).

$$z = y_o + ax + by \qquad \text{(Eq. 3.5-1)}$$

[3] Results from TSPA-SR were not available for the first iteration, but can be used in subsequent iterations.

The regression coefficients are summarized in *In-Package Chemistry Abstraction* (CRWMS M&O 2000 [129287]) and tabulated in Tables 3.5-2 and 3.5-3.

Table 3.5-2. Response Surface Parameters of pH for Commercial Spent Nuclear Fuel Packages

$z=y_o + ax + by$ (see footnote)					
TSPA-SR Parameter	**Response Surface**	**y_o**	**a**	**b**	**Conditions**
PH_CSNF	1	3.4916	-1.0918	0.4571	Early time and low corrosion rate
PH_CSNF	2	3.3977	-0.7468	0.3515	Early time and high corrosion rate
PH_CSNF	3	6.0668	-0.5395	4.0479	Late time and low corrosion rate
PH_CSNF	4	6.0913	-0.3057	1.2913	Late time and high corrosion rate

Source: CRWMS M&O 2000 [148050], Table 4.6

NOTE: z = pH; $x = \log_{10}$(fraction of CSNF matrix covered by cladding), y_o = pH at zero water flux and small fraction of matrix covering and y = water flux (m^3/yr)

Table 3.5-3. Response Surface Parameters of pH for Co-disposal Waste Packages

$z=y_o + b'y + cu$ (see footnote)					
TSPA-SR Parameter	**Response Surface**	**y_o**	**b′**	**c**	**Conditions**
pH_Codisposal Waste Package	5	5.1257	2.6692	0.0764	Early time and low corrosion rate
pH_Codisposal Waste Package	6	4.7324	0.7307	0.0837	Early time and high corrosion rate
pH_Codisposal Waste Package	7	8.4247	-3.4173	0.1403	Late time and low corrosion rate
pH_Codisposal Waste Package	8	9.2554	-3.1280	-0.0418	Late time and high corrosion rate

Source: CRWMS M&O 2000 [148050], Table 4.11

NOTE: z = pH; y = water flux (m^3/yr); y_o = pH at zero water flux and small relative glass dissolution rate, and u = relative glass dissolution rate (dimensionless)

For CSNF waste packages, the cladding coverage parameter, x, is calculated in the cladding degradation subcomponent (Section 3.5.4). The water flux rate parameter, y, is calculated in the seepage-into-drifts subcomponent model (Section 3.2.4). The values of these two parameters at high and low corrosion rates (Table 3.5-3) are used to calculate the uncertainty limits for the in-package pH at each time step. To elaborate, one pH surface was generated for a low waste package corrosion rate case, and a second pH surface was generated for a high waste package corrosion rate case. These surfaces constitute the boundaries of the range of pH values for each time period (early and late) and each package type (CSNF and co-disposal waste packages) using Equation 3.5-1. It is assumed that the actual waste package corrosion rate will fall between the low and high values, and, therefore, the in-package pH would be between the boundaries.

For co-disposal waste packages, pH again depends on the water flux rate parameter, y, calculated in the seepage into drifts-model subcomponent model (Section 3.2.4). The relative glass dissolution rate, u, is set at a constant value. Similar to CSNF waste packages, the values of these two parameters at high and low corrosion rates (Table 3.5-3) are used to calculate the uncertainty limits for the in-package pH at each time-step.

The carbonate ion concentration was calculated from equilibrium mass action equations. The carbonate concentration is a function of the partial pressure of CO_2 and pH. With the partial pressure of carbon dioxide (f_{CO2}) fixed at 10^{-3} bars, the equation is reduced to a sole function of pH. The chloride concentration, also based on composition of J-13 water (Harrar et al. 1990 [100814]), is a constant value and equal to 2.014×10^{-4} M, but is not currently used in any other TSPA model components. The partial pressure of oxygen is fixed at the atmospheric value, which is conservative in that it favors higher solubilities. The ionic strength was defined by a distribution and sampled at the beginning of each TSPA-SR realization and held fixed in time. These model abstractions for the important chemical parameters are tabulated in Table 3.5-4.

Table 3.5-4. Distributions and Constants Used by In-Package Chemistry Component

Chemical Parameter	Parameter Distribution or Value	Description
Log of Oxygen Partial Pressure (bar)	-0.7	Atmospheric conditions—conservative estimate
Log of Carbon Dioxide Partial Pressure (bar)	-3.0	Mountain atmospheric conditions
Total Carbonate Concentration (M)	$10^{-4.47}+10^{-10.82}/10^{-pH}+ 10^{-21.15}/10^{-2pH}$	Applicable to commercial spent nuclear fuel waste form degradation; this equation is a direct consequence of fixing the carbon dioxide partial pressure at mountain atmospheric conditions.
Ionic strength in CSNF waste packages at early times (M)	Beta Distribution Min=2.76×10^{-3} Max=2.92×10^{-3} Mean=2.82×10^{-3} Standard Deviation=5.17×10^{-5}	Applicable to Early Time
Ionic strength in CSNF waste packages at late times (M)	Beta Distribution Min=2.83×10^{-3} Max=3.94×10^{-1} Mean=9.48×10^{-2} Standard Deviation=1.302×10^{-1}	Applicable to Late Time
Ionic strength in co-disposal packages at early times (M)	Beta Distribution Min=2.54×10^{-3} Max=3.48×10^{-3} Mean=3.18×10^{-3} Standard Deviation=2.764×10^{-4}	Applicable to Early Time
Ionic strength in co-disposal waste packages at late times (M)	Beta Distribution Min=7.86×10^{-3} Max=1.35 Mean=3.38×10^{-1} Standard Deviation=4.764×10^{-1}	Applicable to Late Time

Source: CRWMS M&O 2000 [148050]

3.5.2.3 Treatment of Uncertainty and Variability

The uncertainty and variability characteristics of the in-package chemistry model abstraction are essentially the same as those of the in-package chemistry process model. The primary sources of uncertainty are associated with: (1) the thermodynamic database used in the EQ3/6 (CRWMS M&O 1998 [102837]) computer code, (2) the kinetic-rate laws and constants for the waste forms and waste package materials, (3) the composition of the inflowing water, and (4) the mixing cell approximation and water flow model.

The thermodynamic database in the EQ3/6 (CRWMS M&O 1998 [102837]) code represents the best available information in the scientific literature on the properties of minerals and complexes. The kinetic relationships used to describe the dissolution of the waste forms and waste package materials have uncertainty because of the limitations of the experimental procedures used in their determination. In developing the model abstraction, attempts were made to explicitly account for these uncertainties by sampling from a range of values or, where possible, assume conservative parameters. Because the in-package chemistry is dominated by the chemical processes inside the waste package, the chemical composition of the water entering the package was assumed to be J-13 well water. Lastly, the conceptual representation of water flow through the failed package is highly simplified (i.e., mixing cell) and assumes a fully flooded condition. Since this latter assumption may be nonconservative for evaluating localized chemistry (e.g., pH) present on the surface of CSNF cladding, the localized corrosion model for cladding does not specifically use the estimated pH and fluoride and chloride concentration. Rather, localized corrosion is only dependent upon the cumulative amount of water that enters a waste package as further explained in Section 3.5.4.2.

The range of potential chemical environments simulated with EQ3/6 (CRWMS M&O 1998 [102837]) code is large and likely bounding since a number of natural processes would tend to prevent extreme chemistries outside the range of the process model simulations. For instance, the substantial reservoirs of freely exchangeable carbon dioxide would tend to prevent excursions to hyperalkaline conditions. In addition, dissolution of solid components in the waste package can buffer pH as well. The fact that free oxygen is likely to prevail in the drifts sets limits on the reduced chemical conditions. Moreover, the accumulation of reactive corrosion products formed during waste package degradation would buffer the solution chemistries.

As discussed elsewhere in the text (e.g., Section 3.3.2 and Figure 3.3-3), spatial variability of environmental water conditions is explicitly modeled in TSPA-SR using a combination of 5 infiltration bins and 3 drip conditions over the 2 types of waste packages. The in-package chemistry model provides a water chemistry as a function of the environmental conditions (i.e., water flux) and waste package type. In turn, five components of the waste form degradation model (CSNF matrix degradation, CSNF cladding degradation, HLW degradation, dissolved radionuclide concentration, and colloidal radionuclide concentration) reflect this variability in in-package chemistry.

3.5.2.4 Results and Interpretation

As previously described, the in-package chemistry was calculated for two types of packages (CSNF and codisposal) and divided into two periods (<1,000 years and >1,000 years) after first perforation of the waste package. Furthermore, the uncertainty for these conditions was based on an uncertainty of the degradation rate of all components (metal and waste) in CSNF, the degradation rate of all components (metal and waste in HLW, and seepage rate of water into a waste package. The calculated pH readily shows these differences between package types, time periods, and uncertainty (Figure 3.5-8). For most of the 100,000-year simulational period, most waste packages do not have water enter the package either because they are not under drips or the breached surface area of the waste package is so small. Consequently, the pH does not vary much. Furthermore, without seepage of water, the pH of the codisposal packages varies less than the CSNF packages. Only when water begins to seep in near the end of the 100,000-year simulation period, does the codisposal package pH vary more. (The maximum variation would be about two orders of magnitude versus one order of magnitude for CSNF package pH).

3.5.3 Waste Form Matrix Degradation Component Abstractions

The waste form matrix abstraction estimates the rates at which the CSNF, DSNF, and HLW forms dissolve as a function of the inflow conditions and in-package chemistry. The abstractions for waste form degradation are based on laboratory data obtained under various flow conditions. The technical bases for the waste form degradation components are documented in three reports: (1) *CSNF Waste Form Degradation: Summary Abstraction* (CRWMS M&O 2000 [136060]), (2) *DSNF and Other Waste Form Degradation Abstraction* (CRWMS M&O 2000 [144164]), and (3) *Defense High Level Waste Glass Degradation* (CRWMS M&O 2000 [143420]).

3.5.3.1 Conceptual Model

Basis for Commercial Spent Nuclear Fuel Degradation Conceptual Model–The conceptual model of CSNF degradation (Figure 3.5-9) relates the CSNF matrix degradation rate to the controlling physical (e.g., temperature) and chemical parameters (e.g., pH and partial pressures of O_2 and CO_2). The mechanisms of cladding degradation are also important to CSNF matrix degradation because cladding failure determines the rate and extent to which the fuel matrix is exposed to inflowing fluids (i.e., air, water vapor, and liquid water). The chemical reactions involved in CSNF matrix degradation are generally well understood, and the approach to modeling the degradation process has been based on deriving empirical fits to laboratory data.

To measure the rate of CSNF degradation, multi-year dissolution experiments, with both fresh and spent fuel, were conducted under saturated and unsaturated conditions. These laboratory experiments consisted of (1) static tests, (2) flow-through reactors, and (3) drip tests. From the many experiments performed, an understanding of the mechanisms of spent fuel dissolution has emerged. The uranium dioxide fuel dissolves to form uranyl ions (UO_2^{+2}) when exposed to mildly oxidizing solutions. The rate of UO_2 oxidation depends on the interaction of specific surface species that control the rate-determining dissolution step. The CSNF degradation rates increase with decreasing pH (at low pH) and with increasing carbonate levels (at high pH), suggesting that the adsorbed protons and (or) carbonate ions control the dissolution reaction under flow-through conditions. Important aqueous species that might also affect dissolution rates are calcium and silicon ions, which can form stable corrosion products with low solubilities.

For alkaline conditions, the intrinsic dissolution rate of UO_{2+x} (x may vary from 0 to 1, depending on the mixture of tetravalent and hexavalent uranium) in CSNF was determined by using experimental data from a single pass flow-through reactor which allows UO_2 dissolution to be measured far from solution saturation (no precipitation of dissolved products). For acidic conditions, the variation of the degradation rate with pH was based on qualified data point and some limited collaborating data. The empirical temperature and oxygen rate dependencies derived for alkaline conditions were assumed to be the same as for acid conditions and the dependency on $[CO_3]_T$ was assumed to be insignificant (see also Table 3.5-5 in Section 3.5.3.2). The later assumption is consistent with observations that, under alkaline conditions, carbonate and oxygen control the degradation rate, whereas under acidic conditions, pH is a more significant factor (since protons dominate the surface reaction sites), and the carbonate ion is less significant.

Basis for U.S. Department of Energy-Owned Spent Nuclear Fuel Degradation Model–The DSNF degradation subcomponent model determines the rate of degradation of the DSNF waste category and of the immobilized plutonium ceramic waste (Figure 3.5-10). Because of the large number of distinct waste groups in the DSNF category, the conduct of individual laboratory experiments for each group was not feasible. Consequently, a limited number of DSNF groups were selected to bound the behavior of the DSNF category. The DOE Office of Civilian Radioactive Waste Management and the DOE National Spent Nuclear Fuel Program have collaborated in the identification of DSNF groups that encompass all DSNF waste types for criticality, design-basis events, and TSPA-SR analyses:

- Group 1–Naval SNF
- Group 2–Plutonium/Uranium alloy
- Group 3–Plutonium/Uranium carbide
- Group 4–Mixed Oxide fuel and Plutonium oxide
- Group 5–Thorium/Uranium carbide
- Group 6–Thorium/Uranium oxide
- Group 7–Uranium-metal
- Group 8–Uranium oxide
- Group 9–Aluminum-based SNF
- Group 10–Uranium Nitride SNF (with Unknown matrix)
- Group 11–Uranium-Zirconium-Hydride.

The first group of DSNF, consists of 65 MTHM of naval SNF. Each of the naval SNF canisters will be packaged in one waste package, for a total of 300 waste packages. The Naval Nuclear Propulsion Program modeled the performance of naval SNF with the same environmental conditions used for CSNF. Because of its robust cladding design (see Section 3.3.1) releases from naval SNF are very small (Mowbray 2000 [152077]). More importantly, a comparison of releases from naval SNF and CSNF shows that it is very conservative to represent releases from naval SNF with releases from CSNF (see Appendix G).

For each of the other ten DSNF groups and the ceramic plutonium waste form, three types of degradation models were initially developed that consisted of: (1) an upper-limit model, (2) a conservative model, and (3) a best estimate model. The upper-limit model represented essentially instantaneous dissolution of the waste form following exposure to groundwater. The conservative model was derived using only the high-dissolution rates. The best-estimate model was based on consideration of all available laboratory data. To simplify the model abstraction, a single, constant DSNF matrix degradation rate was used to bound Groups 2 through 11 of the DSNF types. The N-Reactor DSNF "conservative model" was selected as the basis for this rate because the N-Reactor model exhibited dissolution rates greater than other groups, and because of the relatively large database on N-Reactor fuel behavior.

Basis for High-Level Radioactive Waste Glass Degradation Model–The purpose of the HLW glass degradation subcomponent is to describe borosilicate glass degradation for the range of conditions (immersion, humid air, and dripping water) to which it is likely to be exposed after waste package failure. The rate of radionuclide release from HLW is calculated by multiplying the glass degradation rate by the mass fraction of the radionuclide in the glass (Figure 3.5-11). This approach for calculating the radionuclide release rate is based on the two assumptions that full immersion degradation rate is bounding and that the release of radionuclides is congruent with (or, alternatively, proportional to) the degradation rate of the borosilicate glass.

Concerning the latter assumption, because the release of radionuclides from the glass will depend on the prior dissolution of the glass, the dissolution rate of the glass imposes an upper bound on the radionuclide release rate.

Concerning the former assumption, degradation of borosilicate glass is assumed to occur as if the glass were fully immersed in water, although it is expected that much of the glass would, instead, be exposed to humid air or dripping water conditions. This assumption is based on a comparison of a model that was developed for immersion with the model for glass degradation in humid air or dripping groundwater conditions. This comparison showed that the rate of glass corrosion under humid air and dripping water conditions was conservatively bounded by the dissolution rate under immersion conditions (CRWMS M&O 2000 [143420]). This assumption is consistent with the conceptual model for in-package chemistry.

3.5.3.2 Implementation in the Total System Performance Assessment

Commercial Spent Nuclear Fuel Matrix Degradation–The CSNF dissolution rate abstraction (CRWMS M&O 2000 [136060], Section 7) is based on in-package chemical conditions (Figure 3.5-9); specifically, pH, temperature, total carbonate (CO_3) concentration, and the partial pressure of oxygen (O_2). The intrinsic dissolution rate, k_o (mg/m^2-day), is defined by the following equation:

$$\mathrm{Log}_{10}(k_o) = a_0 + a_1 / T_k + a_2\, p[CO_3]_T + a_3\, pf_{O2} + a_4\, pH \qquad \text{(Eq. 3.5-2)}$$

where T_k is the waste package temperature in Kelvin, $p[CO_3]_T$ is $-\log_{10}$ of the molar concentration of total carbonate, pf_{O2} is $-\log_{10}$ of the partial pressure of O_2 in bars, and pH applies to fluid inside the CSNF waste package. The coefficient values for Equation 3.5-2 depend on pH reflecting alkaline or acidic conditions. The applicable coefficient values are summarized in Table 3.5-5.

Table 3.5-5. Commercial Spent Nuclear Fuel Intrinsic Dissolution Rate Equation Coefficients as a Function of pH

In-package pH	Coefficients for Commercial Spent Nuclear Fuel Intrinsic Dissolution Rate Equation				
	a_0	a_1	a_2	a_3	a_4
pH>7	4.69	-1085	-0.12	-0.32	0
pH≤7	7.13	-1085	0	-0.32	-0.41

Source: CRWMS M&O 2000 [136060], Section 7

Within the TSPA-SR calculations, the values for total carbonate $[CO_3]_T$ concentration and pH are calculated in the in-package chemistry model, while the partial pressure of oxygen(f_{O2}) is set at constant atmospheric conditions. The waste package temperature is calculated in the thermal hydrology model (Section 3.3.3). In the computer implementation, an uncertainty term was added to a_0 that consists of a uniform distribution with a minimum of –1 and a maximum of +1. This uncertainty conservatively bounds the observed variation of the test results of about 18 percent as noted in *Clad Degradation—Summary and Abstraction* (CRWMS M&O 2000 [147210]) Waste package temperature and in-package chemistry change with time, so the intrinsic dissolution rate is evaluated during each time-step of any given TSPA-SR realization.

U.S. Department of Energy-Owned Spent Nuclear Fuel Matrix Degradation–As recommended in *DSNF and Other Waste Form Degradation Abstraction* (CRWMS M&O 2000 [144164]), the degradation rate of the DSNF is set at a constant value of 10 times the best estimate for the matrix dissolution rate of N-Reactor fuel (i.e., the "conservative model" is 10×0.175 kg/m^2-d). The specific surface area was the area estimated for N-Reactor fuel (7.0×10^{-8} m^2/kg) (CRWMS M&O 2000 [144164], Section 6.4.1).

High-Level Radioactive Waste Matrix Degradation–The HLW glass degradation model is implemented in the form of an analytical expression containing parameters that account for the pH, temperature, surface area, and the combined effects of glass composition and solution composition on the rate of glass corrosion (Figure 3.5-11). Conservative estimates of the parameter values were derived from the fitting of laboratory data. The model for glass dissolution under immersion is based on the rate expression for aqueous dissolution of borosilicate glass. The rate expression to calculate the dissolution rate of HLW glass in an aqueous solution is given by (CRWMS M&O 2000 [143420]):

$$DR = S_{im}\, k_{eff}\, 10^{\eta\, pH} \exp(-E_a/RT) \qquad \text{(Eq. 3.5-3)}$$

where,

$$k_{eff} = k_o\,(1 - Q/K) \qquad \text{(Eq. 3.5-4)}$$

with the coefficients in the above equations defined as follows:

DR = Dissolution rate of the glass (day^{-1})
S_{im} = Specific surface area of glass immersed in water (m^2/g)
k_{eff} = Effective dissolution rate constant (g/m^2-day)
k_o = Intrinsic dissolution rate (g/m^2-day)
Q = Concentration of dissolved silica (g/m^3)

K = Fitting parameter equal to apparent silica saturation concentration (g/m^3)
η = pH dependence coefficient (dimensionless)
E_a = Effective activation energy (kJ/mol)
R = Universal gas constant (8.31451 kJ/(mol-K))
T = Absolute temperature (K).

By selecting a conservative bound for S_{im} and expressing the uncertainty in Q and K in k_{eff}, the model simplifies to an equation in three parameters (k_{eff}, η, E_a) and two independent variables (pH and T). This simplified equation was used in the Waste Form Degradation Model. Values for pH, and T depend on the exposure conditions which vary with time and location (CRWMS M&O 2000 [148384]). The model parameters k_{eff}, η, and E_a are represented using statistical distributions to account for uncertainty in the bounding values. Different distributions are used at acidic and alkaline pH values because the rate law for high-level waste degradation is "U-shaped" with the minimum near neutral pH (Table 3.5-6). The specific surface area of the glass S_{im} is a constant equal to 5.63 x 10^{-5} m^2/g (CRWMS M&O 2000 [148384], Section 6.3.4.4) and R, the universal gas constant, is as previously defined.

Table 3.5-6. Parameter Distributions Used in Propagating Uncertainty in High-Level Radioactive Waste Glass Dissolution Rates Using Equation 3.5-3

Parameter Name	Description	Parameter Value
$Log(K_{eff})_{high}$	Logarithm of K_{eff} @ alkaline pH	Uniform Distribution - Min=6.4, Max = 7.4
η_{high}	Dependence coefficient @ alkaline pH	Uniform Distribution - Min=0.3, Max=0.5
$(E_a)_{high}$	Logarithm of E_a @ alkaline pH	Uniform Distribution - Min=70, Max=90
$Log(K_{eff})_{low}$	Logarithm of K_{eff} @ acidic pH	Uniform Distribution - Min=8, Max=10
η_{low}	Dependence coefficient @ acidic pH	Uniform Distribution Min=-0.7, Max=-0.54
$(E_a)_{low}$	Logarithm of E_a @ acidic pH	Uniform Distribution - Min=43, Max=73

Source: CRWMS M&O 2000 [143420]

At the initiation of the simulation, the TSPA-SR model samples a value from each of the parameter distributions. The TSPA-SR model then calculates the glass dissolution rates for both high and low pH cases. Once both dissolution rates have been calculated, the model compares the high pH and low pH dissolution rates and selects the larger of the two rates. Finally, the rate per surface area is multiplied by the geometric surface times a factor of 21 multiplier. This factor was based on the estimated maximum amount of fracture-induced surface area caused by thermal and mechanical stress.

3.5.3.3 Treatment of Uncertainty and Variability

Commercial Spent Nuclear Fuel Model–The abstracted CSNF degradation model is based on fitting data from flow-through experiments, in which dissolved material is washed away rapidly, over a wide range of conditions relevant to the potential repository. Because flow-through experiments omit the effects of saturation of the water and formation of secondary minerals that may incorporate a few radionuclides (e.g., ^{237}Np), the CSNF model conservatively omits these effects as well. The omission is not significant in the CSNF model but is significant as concerns the solubility of ^{237}Np, as mentioned again in Section 3.5.5.3. As previously mentioned, the uncertainty of the fitted CSNF degradation model in Equation 3.5-2 was estimated to be about one order of magnitude. This uncertainty was directly included in the TSPA-SR, by sampling from a uniform distribution. This estimate of one order of magnitude is based on the evaluation of the goodness-of-fit and consideration of the uncertainty associated with using data from recently irradiated SNF (less than 30 years out of reactor) and unirradiated UO_2 (CRWMS M&O 2000 [136060]).

This model has been compared to YMP-specific data from relevant laboratory experiments, primarily the unsaturated drip tests and batch tests. Long-term drip testing of CSNF under conditions that mimic geologically unsaturated (i.e., limited water and oxidizing) conditions, has been done over the past six years to determine the relationship between the rate of CSNF alteration (i.e., dissolution and secondary phase formation) and the release rate of radionuclides. The model adequately bounds the spread of values reflected in the available dissolution rate data (CRWMS M&O 2000 [136060]).

In addition, favorable comparisons were observed with information from a relevant natural analog site—Nopal I—a uranium-mining site at Peña Blanca, Mexico (Murphy et al. 1997 [100470]). Overall, the phase assemblage observed at Nopal I is similar to that derived experimentally in the CSNF alteration drip tests. The general agreement between the observed alteration products in the various tests, the natural analogues, and the geochemical modeling, provide confidence that the mechanisms of SNF corrosion are well understood and that the degradation model is bounding for long-term projections of CSNF degradation rates.

U.S. Department of Energy-Owned Spent Nuclear Fuel Model–At present, there is a lack of laboratory data for many of the DSNF waste forms, and only a small amount of dissolution data for those wastes forms have been tested. This absence of data has limited the development of detailed models for DSNF and immobilized-ceramic-plutonium degradation. The selection of a bounding estimate for the DSNF dissolution rate mitigates the inherent uncertainties from the limited experimental data base.

TSPA-SR sensitivity calculations for DSNF indicate that the expected annual dose is not sensitive to the DSNF degradation kinetics (see Section 5.2.4). This insensitivity is largely explained by the fact that the DSNF waste inventory only contributes about 2 percent of the total waste inventory of ^{99}Tc and ^{237}Np. The significance of this insensitivity in the TSPA-SR model means that the propagation of uncertainty and variability in DSNF degradation rates is not important to overall performance and the bounding estimate is adequate for TSPA-SR simulations.

High-Level Radioactive Waste Model–The abstracted model is designed to bound the rate at which borosilicate glass will corrode when immersed in groundwater or exposed to humid air and (or) dripping water in the potential repository. The general algebraic form of the fitted mathematical model is widely accepted and used in the scientific literature (Bourcier 1994 [101563]). As described previously, the primary model parameters (k_{eff}, η, and E_a) are represented using statistical distributions to explicitly account for uncertainty and variability. The distributions for η and E_a were based on single-pass flow though experiments for a simple, five-component analogue glass; the distribution of k_{eff} was based on the short-term Product Consistency Tests (CRWMS M&O 2000 [143420]).

The available evidence suggest that the degradation rate calculated the defined distributions bounds for the forward dissolution rates for the currently proposed waste glass compositions. Projected dissolution rates have also been compared with the dissolution rate of basaltic glass recovered from the seabed. The dissolution rates of several samples were calculated, based on the thickness of an alteration layer and the age of the basalt, under seawater pH conditions. The comparison showed that the rate expression is conservative by about one order of magnitude in projecting the long-term dissolution rate of basaltic glass (CRWMS M&O 2000 [143420]).

3.5.3.4 Results and Interpretation

The degradation rate of DSNF is a bounding constant. Because the inventory of ^{99}Tc and ^{237}Np in DSNF is such a small percentage of the total inventory (~2 percent), and the rate of degradation is so fast relative to the time steps of the TSPA-SR calculations (i.e., 500 yr), the TSPA-SR results are insensitive to the DSNF inventory or degradation rate (see Section 5.2.4).

The degradation rate of HLW glass (expressed as rate times the specific surface area) varies between $\sim 10^{-3}$/yr and $\sim 5 \times 10^{-8}$/yr (Figure 3.5-12). This uncertainty range is large and almost entirely due to the uncertainty assumed for the modeling parameters (k_{eff}, η, and E_a) since the uncertainty in pH has only a small influence when the pH dramatically changes after 1,000 years. Even with the large uncertainty, none of the modeling parameters show up as influencing uncertainty in the dose in the sensitivity studies described in Section 5.1 since the contribution of co-disposal packages to dose is an order of magnitude less than CSNF packages. The calculated degradation rate of glass is a function of temperature, which, in turn, decreases with time; however, the temperature influence is small after 10,000 years when waste packages begin to breach; hence, there is only a slight decrease in the mean degradation rate over time (Figure 3.5-12). For comparison, the glass degradation rate of a simulation using mean values varied between 10^{-2}/yr at times less than 1,000 years and 3×10^{-6}/yr at 10,000 years for the TSPA-VA (CRWMS M&O 1998 [108000]).

The degradation of CSNF behaves similarly to HLW but since the rate is intimately tied to the unzipping rate of the cladding, it is presented below in Section 3.5.4.4.

3.5.4 Cladding Degradation Component Abstraction

The cladding degradation component determines the fraction of fuel rods in the CSNF waste packages with perforated cladding as a function of various failure mechanisms induced by physical and chemical processes. Because these mechanisms vary with time, the rate at which the rods fail (by perforation and unzipping) determines the rate at which the CSNF waste matrix is exposed to water. The technical basis for the cladding degradation model component is summarized in the *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332]) and documented in one primary analysis/model report, *Clad Degradation—Summary and Abstraction* (CRWMS M&O 2000 [147210]). This report is supported by seven other analysis/model reports, the major two of which are *Clad Degradation-FEPs Screening Arguments* (CRWMS M&O 2000 [150099]), and *Initial Cladding Condition* (CRWMS M&O 2000 [148249]).

3.5.4.1 Conceptual Model

Since the 1950s, most CSNF has been clad with a thin layer (usually between 0.6 to 0.9 mm) of Zircaloy, an alloy that is about 98 percent zirconium, with small amounts of tin, iron, niobium, and chromium. Zircaloy cladding is not a designed engineered barrier of the potential Yucca Mountain disposal system, but, rather, is an existing characteristic of the CSNF that will affect the rate of release of radioisotopes once engineered barriers (such as the waste package) have failed. Zircaloy cladding is resistant to corrosion based on considerable material and performance data compiled on Zircaloy cladding over the past 40 years (CRWMS M&O 2000 [151659]). Therefore, additional cladding perforation beyond that initially occurring in the reactor or during dry storage is expected to be minimal in the first 10,000 years, especially if water is not present.

The degradation of CSNF cladding is assumed to proceed through two distinct steps: (1) rod perforation of the cladding through the formation of small cracks or holes and (2) progressive exposure of UO_2 SNF matrix (Figure 3.5-13). Perforation of the cladding may occur because:

- Cladding initially perforates within the reactor or during storage.
- Cladding perforates from creep when in dry storage (or disposal at high temperatures) or stress corrosion cracking (SCC) from high stress when temperatures are 300°C or greater.
- Cladding perforates as a result of ground motion and accelerations induced by an earthquake.
- Cladding perforates from localized corrosion as a result of halogen anions (e.g., fluoride or chloride) inside the waste package.

Other mechanisms of initiating cladding perforations have been examined (e.g., hydride failures, hydride embrittlement of cladding, delayed hydride cracking, water-logged rods, general corrosion of cladding, microbial corrosion of cladding, acid corrosion from radiolysis, and enhanced cladding corrosion from high, dissolved-silica content of waters, or diffusion-controlled cavity growth). These mechanisms, however, were screened out because of their low potential to occur and thereby influence the dose. Perforation is assumed to occur at the center of the rod, because this ensures the fastest complete unzipping of the cladding and exposure of the CSNF waste matrix.

Perforation of the cladding permits the fuel matrix to react with the inflowing moisture and air. This reaction is represented by assuming that the UO_2 fuel matrix forms schoepite, an oxyhydroxide mineral of uranium that has a greater volume than UO_2. The expansion of the fuel matrix, in turn, is assumed to induce further cracking and unzipping of the cladding. This process progressively exposes more of the waste matrix as the unzipping continues. The unzipping of the cladding, called unzipping, is a function of the CSNF waste-matrix alteration.

Initial Cladding Condition–Two aspects of initial cladding condition are important (Figure 3.5-13). The first aspect is the number of rods of CSNF that arrive for disposal with cladding perforations. Zircaloy cladding perforations may occur in the reactor (including fuel handling), in pool storage or dry storage (including fuel handling), and in transportation from storage. The TSPA-SR abstraction is based on data from 65,000 BWR assemblies, with about 4 million fuel rods, and 47,000 PWR assemblies, with about 10 million fuel rods (CRWMS M&O 2000 [148208]). For conservatism, the stainless steel cladding (about 1.1 percent of the CSNF) is assumed perforated at the same time as the waste package fails.

The second aspect is the condition and characteristics of the intact cladding (i.e., the hoop stress from internal gas pressure and strain history of the cladding) in order to analyze the potential perforation of the Zircaloy cladding from creep rupture or stress corrosion cracking (SCC) after disposal. The mechanical state of the fuel rods is conservatively estimated by assuming that all Zircaloy cladding was placed in dry storage, then placed in a shipping cask for three weeks, and reached a temperature of 350°C.

Creep and Stress Corrosion Cracking Perforation–A sampling of 2,000 rods, exposed to specific temperature profiles (including dry storage and shipping conditions, with a projected potential repository temperature history), was used to calculate creep strain. The failure-strain criterion for creep rupture was developed based on tests of unirradiated cladding (CRWMS M&O 2000 [148429]). The criterion developed is very conservative, based on a comparison with failure criteria developed by four different sources. At a peak waste package surface temperature at about 300°C and higher, the fraction of rods perforated from creep rupture increases dramatically.

SCC of Zircaloy requires an aggressive chemical environment and high stress. Zircaloy is not susceptible to SCC in NaCl, HCl, $MgCl_2$, and H_2S solutions, but is in iodine solutions, when the iodine concentration in the fuel-cladding gap is greater than 5 g/cm^2. Free iodine concentrations are expected to be negligibly small in CSNF. However, for the TSPA-SR, it was conservatively assumed that a sufficient amount of iodine was present on the cladding interior, the stress was high, and the duration of elevated temperatures was sufficiently long so that once cracking started, there was sufficient time to propagate through the cladding.

From the statistical expression of the initial internal pressures, temperature history during transportation and storage, and the temperatures during disposal, the perforations from creep and SCC were calculated, and a table of cladding perforation from creep rupture and SCC at various temperatures was produced. Estimation of postdisposal cladding temperature is based on a heat conduction model. The internal temperature is evaluated as a function of waste package surface temperature, and so internal temperatures, can also be readily calculated.

Localized Corrosion and Cladding Perforation–Most known forms of localized corrosion of the cladding could be screened out based on the calculated macroscopic, in-package chemistry. For example, fluoride, which is present in Yucca Mountain groundwater in small concentrations (2.2 mg/L in the J-13 well [Harrar et al. 1990 [100814]]), may corrode Zircaloy if the concentration is sufficiently high and the pH is sufficiently low. However, the fluoride concentration is too low by several orders of magnitude at Yucca Mountain to corrode Zircaloy. Yet localized corrosion was still considered as a perforation mechanism, albeit generically, to account for uncertainty in the microscopic local in-package chemistry. The localized corrosion was assumed to be linearly dependent upon the cumulative amount of an anion (such as fluoride) that had entered a waste package.

Mechanical- and Seismic-Induced Cladding Perforation–Perforation of the Zircaloy cladding may also arise from mechanical and seismic loads. Rockfalls, and even most earthquakes, would not perforate the CSNF cladding. Only severe seismic events, with a frequency of 1.1×10^{-6}/yr could potentially perforate the cladding. A disruptive event with this frequency is considered in the TSPA-SR simulation. This event is assumed to cause the perforation of all cladding, making it susceptible to further degradation by unzipping.

Cladding Unzipping–Unzipping of the cladding occurs after perforation. The release occurs in two stages: fast release and wet unzipping. Fast release refers to the inventory of radionuclides that are in the gap between the fuel pellets and the cladding and the radionuclides at the grain boundaries of the SNF matrix (Section 3.5.1) plus the inventory in the matrix at the perforation for a specified volume.

In the second stage of release, wet unzipping is assumed to occur as dissolved UO_2 precipitates locally as schoepite. This results in a volume increase, which could hypothetically induce tears and unzipping ("splitting") of the cladding. Wet unzipping at the potential repository site is not expected, because it has not been observed in reactor storage pools. However, because of the inherent uncertainty, and because wet unzipping conservatively bounds diffusive releases of radionuclides out of the perforation pinholes, it is modeled. The cladding component models the unzipping rate as equal to the intrinsic dissolution rate of the CSNF evaluated in the CSNF matrix degradation model component times an enhancement factor sampled from a triangular distribution with a mode of 40 and range of 1 to 240. Because the CSNF degradation rate varies with pH, total carbonate concentration, and temperature, the unzipping velocity and fraction of fuel exposed are evaluated at each time-step in TSPA-SR. Rapid oxidation of CSNF UO_2 to U_3O_8 under dry conditions ("dry unzipping") is unlikely to occur for most CSNF cladding, because the waste package is expected to last far beyond 200 years, and, therefore, fuel temperatures would be too low at time of cladding perforation for the phenomenon to be feasible. Furthermore, the incubation period is quite long for dry unzipping such that wet unzipping rate bounds the dry unzipping rate for cladding that is emplaced initially perforated.

3.5.4.2 Implementation in the Total System Performance Assessment

Initial Cladding Perforation–To define the percent of fuel rods that are perforated before emplacement in the potential repository, the *Initial Cladding Condition* (CRWMS M&O 2000 [148249]) and *Clad-Degradation—Summary and Abstraction* (CRWMS M&O 2000 [148208]) develop a triangular distribution, with the minimum equal to 0.0155 percent, the mode to 0.0948 percent, and the maximum equal to 1.285 percent. This distribution for cladding perforation is based on projected burnup of the fuel to 75 GWD/MTU (gigawatt day per metric tonne of uranium). This distribution is sampled once for each simulation to provide the initial failure percentage for the fuel rods for all waste packages in all five infiltration bins of waste packages defined in the TSPA-SR (see Section 3.3.2 and Figure 3.3-3). Thus, these initially perforated rods are immediately available for cladding unzipping when the waste package fails.

Creep and Stress Corrosion Cracking Failures–The report entitled *Clad Degradation—Summary and Abstractions* (CRWMS M&O 2000 [148208]) presents a model for cladding perforation as a function of creep and SCC (CRWMS M&O 2000 [147210], Section 6.2). At any given waste package temperature, the lower, mean, and upper fraction of perforated cladding is specified. In the TSPA-SR, the peak waste package temperature that occurred during disposal in each infiltration bin is used to select the lower, mean, and upper fraction. The model then samples from these values, assuming a triangular distribution in order to determine the fraction of rods perforated by creep rupture and SCC (Table 3.5-7). Because the waste package surface temperature rarely exceeds 200°C, most of the data in Table 3.5-7 is not used. Only the first two rows are necessary. The creep rupture and SCC specified in the first row is primarily because of the assumed dry storage conditions. The mean of the distribution for the most commonly used first row is 0.075 or about 8 percent. Thus, on average about 8 percent of the cladding will be calculated to perforate because of creep rupture or SCC. This mean percentage is very conservative and likely above the amount of creep and SCC that the NRC will tolerate of operators of dry storage facilities.

Table 3.5-7. Fraction of Rods Perforated from Creep and Stress Corrosion Cracking as a Function of Peak Waste Package Surface Temperature

Peak Waste Package Temperature (C)	Lower Limit	Mode	Upper Limit
≤ 177	0.0105	0.0244	0.1942
227	0.0105	0.0244	0.1949
252	0.0105	0.0258	0.2057
262	0.0105	0.0267	0.2156
277	0.0106	0.0339	0.2479
292	0.0120	0.0604	0.3264
297	0.0133	0.0783	0.3628
302	0.0173	0.0987	0.4080
312	0.0370	0.1622	0.5052
327	0.1067	0.3019	0.6379
352	0.3424	0.5567	0.8227
377	0.5920	0.7789	0.9553
402	0.7986	0.9302	0.9970
≥ 412	0.8720	0.9658	0.9985

Source: CRWMS M&O 2000 [147210]

Localized Corrosion–After failure of the waste package, the fraction perforated by localized corrosion from fluoride is proportional to the volume of water that has entered the package. The fraction is one when 2,424 m^3 of water has entered the waste package (CRWMS M&O 2000 [148208]). This analysis makes the rod perforation fraction linearly dependent on the cumulative amount of water entering the waste package. The uncertainty range is defined in the model as a log-uniform distribution ranging from 0.1 to 10. The water flow depends on the location of the waste package because of different drip rates in different infiltration bins of the potential repository. The water flow into the waste package increases with time as additional patches open on the waste package.

Stainless Steel Cladding–A small number of the CSNF waste packages will contain stainless steel clad fuel. For these CNSF waste packages, the assumption is that all the stainless steel clad fuel is initially perforated, and, thus, immediately available for unzipping upon waste package failure. This assumption is implemented in the TSPA-SR model by adding the fraction of stainless steel clad fuel to the fraction of Zircaloy cladding that is initially perforated. Other than this difference, the radionuclide release from these packages is modeled like the CSNF waste packages. In TSPA-VA (CRWMS M&O 1998 [108000]), the 1.1 percent of stainless steel clad fuel was assumed to be uniformly placed in each and every CSNF waste package, which resulted in only a small portion of a rod being placed in a waste package. For TSPA-SR, a more realistic loading arrangement was assumed. About 3.49 percent of the containers were assumed to be 32.9 percent filled with stainless steel clad fuel. Since the number of waste packages containing stainless steel cladding is small, these packages are not distributed evenly among the environments bins. Rather, they are assigned to the bin covering the largest area of the repository. For the low infiltration case, the largest bin is the first bin. For the mean and high filtration cases, this is the fourth bin. Furthermore, the stainless steel clad fuel is assigned to the "always dripping" condition; consequently, this fuel contributes both to advective and diffusive releases.

Seismically Induced Cladding Failure–Seismic perforation of CSNF cladding has been implemented as a discrete event (event generator in GoldSim) with a frequency of occurrence of 1.1×10^{-6}/yr (CRWMS M&O 2000 [148249], Section 6.4.1). When this event occurs in the TSPA-SR simulation of the nominal scenario, the CSNF cladding in all waste packages is assumed to perforate and unzipping is initiated when the waste packages fail.

Clad Unzipping–This occurs to fuel rods that are in failed waste packages and have perforated cladding which may be degrading further by the process of unzipping. As already discussed, the exposed fuel matrix is assumed to react at the intrinsic dissolution rate and form schoepite. Such alteration results in significant volume expansion, so the cladding unzipping will eventually propagate from its original location to the ends of the active length. It is conservatively assumed that a perforation occurs in the center of a fuel rod and propagates in both directions to the ends of the rod.

The time it takes a rod to unzip is one-half the length of a CSNF rod ($1/2 \times 3.66$ m) divided by the unzipping velocity. The unzipping velocity is calculated in proportion to the CSNF waste matrix degradation rate (described in Section 3.5.3.) The report *Clad Degradation—Summary and Abstraction* (CRWMS M&O 2000 [148208]), presents various estimates for the ratio of the unzipping velocity to the intrinsic dissolution rate and, as already mentioned earlier, proposes a triangular distribution (minimum = 1, mode = 40, and maximum = 240). In the TSPA-SR simulation, a sampled value from this distribution is multiplied by the CSNF waste matrix degradation rate to determine the unzipping velocity.

Average Commercial Spent Nuclear Fuel Waste Matrix Exposed–The pH within the package is a function of the fraction of the CSNF waste matrix that is exposed (Section 3.5.2). However, because of computational limitations, individual packages within the potential repository are generally grouped into five infiltration bins and three dripping conditions to which they are exposed (see Section 3.3.2 and Figure 3.3-3). Hence, an average fraction of CSNF waste matrix is exposed (alternately, a fraction of cladding unzipped and failed) for each group.

The amount of radionuclides that are exposed, or available for dissolution and transport out of the package, is used to calculate the fraction of radionuclides released at any point in time. A long-lived radionuclide (^{238}U), was chosen for evaluating the amount of radionuclides available for transport to ensure the radionuclide has not completely decayed by the end of the calculation. This inventory is adjusted to account for radioactive decay and in-growth via decay of ^{242}Pu. The total amount of ^{238}U within a group is calculated by multiplying the inventory per package by the total number of packages in the group. The amount of ^{238}U within the failed waste packages is then calculated as the number of failed waste packages times the inventory of ^{238}U per package divided by the waste packages per group. This is the amount of ^{238}U that will be exposed once the cladding protection is lost. The ratio between the amount of ^{238}U that is exposed because of both package and cladding failures to the amount of ^{238}U contained within the failed packages gives the average cladding failed at any point in time. This amount is used as input for calculating the pH in the in-package chemistry subcomponent model.

3.5.4.3 Treatment of Uncertainty and Variability

One limitation to this component is that, by necessity, it is based on current fuel and cladding characteristics. Consequently, it is only directly applicable to commercial PWR fuel with Zircaloy cladding and fuel subjected to normal operations at generator sites. Also, fuel burn-up projections used have been limited to the current licensing rules, which include restrictions on fuel enrichment, oxide-coating thickness, and rod plenum pressures. However, ranges of uncertainties have been established, and bounds on these uncertainties were used in developing the model abstraction.

Uncertainty and variability in four model parameters are explicitly accounted for using statistical distributions. These model parameters are: (1) initially perforated cladding, (2) creep and SCC perforation as a function of peak waste package temperature, (3) localized corrosion uncertainty, and (4) unzipping velocity uncertainty. These distributions are based on extensive experiments and observations of cladding behavior made over a 40-year period, as described in *Clad Degradation—Summary and Abstraction* (CRWMS M&O 2000 [148208]). For example, the analysis of initial cladding conditions is based on reactor fuel performance reports that have been published since the start of the nuclear reactor industry. Continuous corrosion experiments with zirconium alloys under low ionic-strength conditions have been performed for nearly three decades. Furthermore, the performance of Zircaloy in boiling seawater and geothermal solutions has been evaluated. Fuel has been exposed in SNF pools for over 25 years and in dry-storage research programs and although only information from PWR clad fuel was used, the performance of BWR cladding is adequately bounded because the cladding on BWR fuel is much thicker. Finally, the analysis of creep and SCC is supported by an extensive experimental basis.

3.5.4.4 Results and Interpretation

Until several tens of thousands of years after the waste package begins to fail, the only cladding perforated is that CSNF cladding that arrives at the site perforated or perforates because of creep rupture during the first few hundred years. For example, the mean perforation is respectively 0.0045 and 0.0765 for initial perforation and creep rupture for infiltration Bin 4 (20 to 60 mm/yr). Thus, the mean perforation is primarily from creep rupture or SCC and approximately equal to the mean of the distribution of the first row of Table 3.5-7. Only after 50,000 years, does the perforated fraction of cladding change due to localized corrosion in those waste packages that have seepage (Figure 3.5-14). The localized corrosion is a direct function of the seepage volume into the waste package and the intermittent drip case, as noted in Section 4, has the greater mean seepage volume; thus, the intermittent drip case has slightly more localized corrosion and greater perforation. More importantly, however, even the very conservative assumption of relating perforation from localized corrosion to the cumulative water volume into the waste package to bound the potential uncertainty from localized corrosion is not influential in the first 100,000 years. Specifically, because cladding perforation is predominately from creep rupture (from dry storage) during the first 40,000 years, the dose is not strongly affected by changing other parameters that influence degradation of the cladding as discussed in Section 5.3.4.

The unzipping of the cladding is assumed to be caused by the alteration of the CSNF matrix; hence, as the cladding unzips the CSNF matrix is exposed. As described above, the unzipping velocity is between 1 and 240 times the degradation rate of the CSNF matrix. This large uncertainty is an important source of the uncertainty in the dose results after 100,000 years but not prior to 100,000 years as discussed in Section 5.1. The velocity (expressed as velocity divided by rod length) varies between 4×10^{-6}/yr to 2×10^{-3}/yr (Figure 3.5-15). For comparison, the CSNF matrix degradation rate had less variation in TSPA-VA (CRWMS M&O 1998 [108000]); the rate varied between 4×10^{-5}/yr and 2×10^{-4}/yr. In the TSPA-VA, perforation of the cladding from rock fall was the primary mechanism to damage cladding beyond 100,000 years whereas in TSPA-SR, conservative assumptions about localized corrosion perforate cladding beyond 100,000 years. In TSPA-SR, with backfill in the design, perforation from rock fall was screened out as a process because of the presence of the drip shields and backfill in the tunnels.

3.5.5 Dissolved Radionuclide Concentration Component Abstraction

The dissolved radionuclide concentration component provides the radionuclide solubilities that are used in the release calculations. The technical basis for the dissolved radioisotope concentration model is documented in *Summary of Dissolved Concentration Limits* (CRWMS M&O 2000 [143569]) and summarized in *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332]).

3.5.5.1 Conceptual Model

Identification or designation of a solubility-controlling phase is of key importance, because the result can affect the calculated radionuclide concentrations by orders of magnitude. In nature, the controlling phase may either be a pure radionuclide solid phase, with the radionuclide as a dominant element, or a solid phase with trace amounts of the radionuclide, as can occur with coprecipitation. For TSPA-SR analysis, only pure phases were chosen because, in general, they yield higher dissolved concentrations. The specific pure phase chosen, in order of preference, was based upon (1) geologic field observations, (2) experimental observation, or (3) crystallochemical arguments. For unnaturally occurring radioelements, such as plutonium and neptunium, experimental observations were relied upon. In those cases where information could not be obtained from field observations or experimental studies, crystallochemical arguments were used. Specifically, it was conservatively assumed that the most amorphous and hydrated (hence, most soluble[4]) oxide of the particular radioelement forms were present.

The dissolved concentration limits conceptual model builds upon three bits of information: (1) estimates of in-package water chemistry (pH, Eh or oxygen partial pressure, ionic strength, carbonate concentration or partial pressure of carbon dioxide), (2) a determination of the likely solubility-controlling phases for the specific radionuclides of concern, and (3) measured (or estimated) thermodynamic parameters describing the stability of aqueous species and solid radioisotope phases. From this information either a functional relationship, a distribution, or a constant value was selected for the radionuclides of importance (Table 3.5-8).

Dissolved Concentrations for Uranium, Neptunium, and Americium–Reactions affecting the solubility of uranium, neptunium, and americium, are primarily carbonate complexation and hydrolysis. Because of this known influence and because enough thermodynamic data were available, the solubility functions of uranium, neptunium, and americium were developed with a dependence on pH and partial pressure of carbon dioxide (f_{CO2}) (bar). Furthermore, the solubility of uranium was also assumed to be a function of temperature (T) (^oC) (Table 3.5-8).

Table 3.5-8. Parameter Values for Total System Performance Assessment Radionuclide Solubilities

Description	Solubility Distribution, Function, or Bounding Value
Logarithm of Americium Solubility (mg/L)	$58.0335-18.9422*pH+1.4744*pH^2-6.0032\log f_{CO2}-0.7005*(\log f_{CO2})^2+0.1162*pH^2\times\log f_{CO2}+ 0.1146*pH\times(\log f_{CO2})^2$
Carbon Solubility (mg/L)	1.2×10^4
Cesium Solubility (mg/L)	1.33×10^5
Iodine Solubility (mg/L)	1.27×10^5
Neptunium Solubility (mg/L)	$7.538 \times 10^{-8} + 1.086 \times 10^{(8-pH)}$
Logarithm of Protactinium Solubility (mg/L)	Log-Uniform Distribution - Min=-4.64, Max=0.36
Logarithm of Plutonium Solubility (mg/L)	Log-Uniform Distribution - Min=-4.62, Max=1.68
Radium Solubility (mg/L)	0.52
Strontium Solubility (mg/L)	8.76×10^4
Technetium Solubility (mg/L)	9.89×10^4
Thorium Solubility (mg/L)	2.32
Logarithm of Uranium Solubility (mg/L)	$7.9946–2.6963pH +0.4292pH^2 – 1.6286\log f_{CO2}+ 0.0095T+0.4161pH\times\log f_{CO2}–0.0051pH\times T–0.0022\log f_{CO2}\times T$ for T <90°C; For T > 90°C, T set to 90°C in equation

Source: CRWMS M&O 2000 [143569]

For uranium solubility, the controlling mineral was assumed to be schoepite, because 1) it is the first mineral to be formed during SNF corrosion; 2) field observations show that it can persist for more than 10,000 years; 3) it has relatively high solubility; and 4) the solubility-temperature relationship is known. For neptunium, the relatively soluble Np_2O_5 was assumed to be the solubility-controlling minerals. For americium, $AmOHCO_3$ was chosen as the conservative solubility-controlling phase.

[4] Although the rule of thumb that the more hydrated phase is the most solubile is not always valid, no exception was known for the radioelements considered here.

Dissolved Concentrations for Plutonium and Protactinium–The solubility of plutonium, protactinium, and lead was represented by a distribution. A value was sampled from the distribution (see Table 3.5-8) and used for the entire simulation. To develop statistical distributions, solubility calculations were generally performed for the range of pH, Eh, and partial pressure of carbon dioxide (f_{CO2}) evaluated by the in-package chemistry model component. For example, for plutonium, solubility calculations were performed with pH in half units from 4 to 8, Eh (V) of 0.34, 0.55, and 0.76 V, and (f_{CO2}) of $10^{-3.0}$ and $10^{-3.5}$ bar.

The primary candidates for solubility-controlling plutonium phases are crystalline PuO_2 and amorphous $Pu(OH)_4$(am) (i.e., $PuO_2.xH_2O$, with x varying from 0 to 2). Since the crystalline phase forms within laboratory time scale, it is reasonable to assume that, over geological time, the less soluble crystalline form, PuO_2, would control plutonium dissolved concentration. However, recent work on the reaction of PuO_2 with oxygenated water (see discussion in CRWMS M&O 2000 [138332]) has found that PuO_2 may be converted into more soluble PuO_{2+x}. Furthermore, because of the potential for some crystalline damage caused by the decay of plutonium isotopes, $Pu(OH)_4$ (am) was used as the controlling solid to provide a more conservative approach.

Few thermodynamic data exist for protactinium. For TSPA-SR calculations, a solubility distribution was based on literature values. The distribution used was identical to that obtained by the informal project expert elicitation used for past TSPAs (Wilson et al. 1994 [100191], Table 9-2b).

Dissolved Concentrations for Technetium, Iodine, Cesium, Carbon, Strontium, and Thorium–Constant bounding values, in which the same bounding value was used for all time steps in all simulations, are used for many of the radionuclides (see Table 3.5-8). This is often done where few thermodynamic data are available or because of low inventory.

Under oxidizing potential repository conditions, no solids are projected to form to limit the solubility of technetium, iodine, cesium, and carbon. Consequently, the solubility of each is set to 1.0 M (mol/L) (Table 3.5-8 converts this concentration to units of mg/L). This high solubility generally means that the degradation rate, diffusion out of the waste package, or the inventory control the respective release. The most likely strontium-bearing solids to precipitate under the potential repository conditions are carbonate or sulfate phases. In lieu of a more involved treatment of sulfate and carbonate levels in the in-package water, the strontium solubility is set to 1.0 M (mol/L). A description of the treatment for the remaining radionuclides is documented in *Pure Phase Solubility Limits-LANL* (CRWMS M&O 2000 [148205]).

3.5.5.2 Implementation in the Total System Performance Assessment

The solubilities of uranium, neptunium, and americium are functions of temperature, pH, and (or) carbon dioxide partial pressure. Temperature and pH vary spatially within the potential repository (see the sections on thermal hydrology and in-package chemistry). Therefore, the solubility of uranium, neptunium and americium also vary spatially within the potential repository according to the environment (i.e., each infiltration bin and each drip condition [see Section 3.3.2 and Figure 3.3-3]).

For example, the solubility of uranium is a function of the in-package pH, the waste package temperature, and the fugacity of carbon dioxide. The solubility of neptunium is a function of the in-package pH. The solubility of americium is a function of the in-package pH and the partial pressure of carbon dioxide. These solubility functions are used both inside the waste package and in the invert. The in-package pH is localized to an environment (i.e., infiltration bin and drip condition). Because the pH in an environment varies with time, all pH-dependent properties will also vary with time, including the solubilities of uranium, neptunium, and americium. The temperature (T) of a waste package is also localized to an infiltration bin. The temperature is a function of the infiltration scenario and time, and, hence, all temperature-dependent properties vary with time, including the solubility of uranium. As previously explained, the carbon dioxide partial pressure is set at the mountain atmospheric conditions for all simulations. The solubilities of uranium, neptunium, and americium are calculated the same way for each bin environment for both waste package types. Hence, the uranium, neptunium and americium solubilities are calculated in each of the 15 environments for each package type.

The solubility of radioelements defined by distributions are sampled at the beginning of each simulation and assumed constant for the entire simulation. The solubility of radioelements defined by a constant is the same at all times for every simulation. Furthermore, the sampled value or constant is used to define the solubility both inside the waste package and in the invert.

In all cases, if multiple isotopes of a radionuclide are present in the aqueous phase, the concentrations of each isotopes is determined by the mass fraction of the isotope present at each time step such that concentration of all isotopes does not exceed the solubility limit of the radioelement.

The solubility of the radioelement is compared to the mass of the element that was available during the time step to determine whether the degradation rate or depletion of the inventory should control the concentration of the radioelement inside the package. To be conservative, the volume of water used for calculating the concentration is only the water held in the pores of the degraded waste matrix not the volume of water in might be present in the entire waste package (Figure 3.5-16).

3.5.5.3 Treatment of Uncertainty and Variability

As discussed elsewhere in the text (e.g., Section 3.3.2 and Figure 3.3-3) and most recently in Section 3.5.5.2, spatial variability of environmental water conditions is explicitly modeled in TSPA-SR using a combination of 5 infiltration bins and 3 drip conditions over the 2 types of waste packages. The in-package chemistry model determines the water chemistry as a function of the environmental conditions (i.e., water flux) and waste package type. In turn, the solubility for neptunium, americium, and uranium) directly reflect this variability in in-package chemistry. For those radionuclides for which a distribution (e.g., plutonium) or bounding constant (technetium) was developed, the ranges of water chemistry observed in the process modeling of in-package chemistry were used when possible; thus, the uncertainty in chemical conditions are indirectly reflected in the distribution or bounding constant. However, the starting composition of major elements in the water was assumed to be J-13 well water and neglected the possibility of high ionic strength compositions from previous evaporation. This high ionic strength condition for the water entering the drift will only occur for a short time after the waste packages cool. The waste packages do not breach until after 10,000 years at which time ionic strength will be closer to ambient conditions. Furthermore, although simulations at high ionic strengths (greater than about 0.7 mol/kg) encounter inherent limitations within the databases, this uncertainty is only up to a factor of two, which is very small relative to other uncertainties within the system. Solubility values for radionuclides, determined by the informal elicitation of expert judgments conducted in 1993 (Wilson et al. 1994 [100191], Table 9-2b), corroborate the newly evaluated distributions and fixed values.

A review of thermodynamic data and controlling phases was performed for a large range of chemical conditions. When uncertainties were encountered, choices were made that would result in higher solubilities. For example and already mentioned, neptunium solubility was conservatively assumed to be controlled by Np_2O_5. Incorporation of the neptunium into secondary phases of uranium such that the solubility of uranium controls the solubility of neptunium is not considered. This assumption is potentially a significant conservatism. As discussed in TSPA-VA when developing the distribution for neptunium (CRWMS M&O 1998 [108000]), some experiments on the degradation of CSNF fuel matrix show a mean concentration of neptunium as low as 2×10^{-3} mg/L (10^{-8} M), whereas the results discussed below show a long term average solubility of 2×10^{1} mg/L (i.e., four orders of magnitude greater solubility). While the effect on dose in the first 10,000 years or even 100,000 years may not be great, the effect of this conservatism on peak dose beyond 100,000 years is important. Consequently, a sensitivity study using an alternative model of several radionuclide solubilities was run and is discussed in Section 4.1.3.

The distributions for the solubility of neptunium americium, plutonium, thorium, and uranium were defined based on high drip rate tests conducted at Argonne National Laboratory. As described by Finn (1994 [100746]), drip tests have been conducted on two CSNF specimens (~8 g). The tests are conducted at 90°C and have been running continuously for six years. The two specimens are exposed to a high drip rate, a low drip, and humid air. From these six tests, 67 water samples have been taken over time and radionuclide concentrations measured. The mean and standard deviation (evaluated on logarithm of the concentration) of five radionuclides from the tests on the two specimens at the high drip rate are given in Table 3.5-9.

Table 3.5-9. Mean and Standard Deviation of Concentration or Radionuclides in High Drip Rate Tests Conducted at Argonne National Laboratory.

Element	Mean Concentration (from logarithm) (mg/L)	Standard Deviation (from logarithm) (mg/L)
Americium	2.52×10^{-2}	1.33×10^{-1}
Neptunium	5.45×10^{-3}	3.08×10^{-2}
Uranium	3.61	1.13×10^{-1}
Plutonium	1.82×10^{-2}	6.97×10^{-2}
Thorium	1.11×10^{-7}	1.94×10^{-7}

Source: CRWMS M&O 2000 [153105]; CRWMS M&O 2000 [131861]

The mean concentration of uranium in the tests is 3.61 mg/L which is near the average solubility of 2.3 mg/L in the CSNF waste packages between 2000 and 40,000 years, prior to much water entering the waste packages (Figure 3.5-17c discussed below). The mean plutonium concentration of 1.82×10^{-2} mg/L in the tests is near the mean solubility of 3.8×10^{-2} mg/L used in the TSPA-SR. However, the mean neptunium concentration of 5.45×10^{-3} mg/L at a mean pH of 6.7 is 3.4 orders of magnitude less than the average solubility of 1.35×10^{1} mg/L at an average pH of 6.8, leading to significantly less release of this radionuclide.

3.5.5.4 Results and Interpretation

As is discussed more thoroughly in Section 4, the most important radionuclides are ^{99}Tc and ^{129}I in the first 40,000 years after waste package failure and ^{239}Pu and ^{237}Np thereafter. Of these radionuclides, ^{99}Tc and ^{129}I have constant solubility values and ^{239}Pu is sampled from a distribution prior to the simulation. Only ^{237}Np solubility is evaluated during the simulation based on the in-package pH. Furthermore, the uncertainty of the in-package pH is assumed to be the sole source of uncertainty of the ^{237}Np solubility. For the CSNF waste packages, the mean solubility of ^{237}Np is 1,000 mg/L in the first 1,000 years, and 20 mg/L (8×10^{-5} M) thereafter. The uncertainty in the pH of the CSNF waste packages is reflected in the spread of the ^{237}Np solubility distribution (Figure 3.5-17a). As explained in Section 3.5.2.4, the uncertainty in the pH of the co-disposal packages is less; consequently, the uncertainty in the ^{237}Np solubility is correspondingly less (Figure 3.5-17b). The distribution for neptunium solubility in TSPA-VA (CRWMS M&O 1998 [108000]), had a mean of 0.34 mg/L (1.4×10^{-6} M) (about two orders of magnitude less than in TSPA-SR). A sensitivity study using a lognormal distribution with a much lower mean solubility for neptunium (5.45×10^{-2} mg/L) is reported in Section 4.1.3.

As described in Section 5.1, the ^{237}Np solubility does not show up as having an important influence on the uncertainty of the dose results in TSPA-SR, whereas in TSPA-VA (CRWMS M&O 1998 [108000]) and TSPA-1995 (CRWMS M&O 1995 [100198]), ^{237}Np was an important parameter. Although the total spread in the uncertainty of ^{237}Np solubility is potentially larger than as used in the TSPA-VA or TSPA-1995, this potential spread in the uncertainty is because of differences in chemistry between CSNF and co-disposal packages, because of the different chemistry between the first 1,000 years after breach and thereafter, and because of the varying water flow rates into the waste packages. In TSPA-SR, the co-disposal packages are not an important contributor to dose, the first 1,000 years is a short period relative to the simulation time, and the pH range is narrow because very little water flows into the package; hence, the ^{237}Np solubility range is narrow and does not influence the uncertainty in the dose.

Although uranium is not a direct contributor to the dose in the first 100,000 years or the peak dose at later times, ^{230}Th, a decay product of ^{234}U is important. The average solubility of uranium in the CSNF packages in the first 100,000 years (2.3 mg/L) (Figure 3.5-17c) is similar to that measured in the high drip rates tests (3.61 mg/L) (Table 3.5-9); however, the solubility of uranium in the co-disposal packages is much greater (Figure 3.5-17d).

3.5.6 Colloidal Radionuclide Concentration Component Abstraction

The function of the colloidal radionuclide concentration component is to calculate the concentration of colloid-associated radionuclides that are generated by the waste forms. Colloid transport is potentially important for radionuclide elements that have low solubility and can be entrained in, or sorbed onto, waste form, engineered barrier, or geologic barrier materials that form colloidal particles. Of these radionuclides, only those that are a major part of the waste inventory and have potentially large dose conversion factors are of potential importance to the performance of the disposal system. The technical basis for the colloidal radioisotope concentration model abstraction is documented in *Waste Form Colloid-Associated Concentrations Limits: Abstraction and Summary* (CRWMS M&O 2000 [148214]).

3.5.6.1 Conceptual Model

Three major types of colloids, based on the source of the colloid substrate material, are recognized to be important for the colloidal radionuclide concentration component (Figure 3.5-18):

1. Waste form colloids formed during degradation of HLW glass (Note: these colloids are further classified into reversibly attached and irreversibly attached radionuclide types.)

2. Corrosion-product colloids formed during corrosion of iron-containing waste packages

3. Groundwater colloids present in the waste form area.

Based on laboratory experiments and data, degradation of borosilicate HLW glass will be the predominant source of waste form colloids. In the case of HLW, the waste form colloids that have been observed consist principally of smectite-type clay minerals. The laboratory data suggest that as HLW glass degrades, colloids are generated and often contain embedded plutonium. Humic substances and microbes were not included in the laboratory experiments, because they are not abundant in YMP groundwater, and because they are typically large enough to be filtered during transport. Relatively few colloids have been observed in laboratory testing of CSNF. In the case of DSNF waste form, there are no experimental data on production of waste form colloids.

The models for all three colloid types assume reversibly attached radionuclides. In addition, the model for colloids from HLW glass includes embedded (or irreversibly attached) americium and plutonium (the latter, as observed in waste glass tests) (Figure 3.5-19). All the model expressions are based on the population of each colloid type (expressed in terms of mass of colloids per volume of fluid) and experimental data for the sorption of radionuclides onto the colloid substrate materials involved. The effects of pH and ionic strength on the stability of dispersions of each colloid type are considered to constrain the estimate of mobile colloid concentrations.

Selection of the radionuclides considered susceptible to colloid-facilitated transport was based on sequential consideration of three criteria. First, candidate radionuclides were identified based on waste inventory and their potential effect on releases. Second, those radionuclides were evaluated considering known transport behavior, based on laboratory and field information. Highly sorbing radionuclides (e.g., Pa and Th) were selected as candidates for colloidal transport to be conservative. Radionuclides known from laboratory work to be embedded in colloids (e.g., Pu and Am) and thus irreversibly attached or suspected to travel as colloids in the field (Pu) were also selected. Third, the major daughters of selected radionuclides were included (Ac, Ra, and Pb).

Concentrations for HLW Colloids–For waste form colloids, the generation and availability of irreversibly attached radionuclides is based on the experimental data for HLW glasses. It was observed experimentally that, as the ionic strength increased, colloid concentration generally decreased, and, ultimately, a threshold value was reached above which the colloids were not readily observed. Hence, at ionic strengths that are relatively low (0.01 mol/kg), colloidal concentration was set in the model to equal the maximum value observed in the experiments (6×10^{-8} mol/L). At moderately high ionic strengths (greater than or equal to 0.05 mol/kg), colloidal concentration was defined as a minimum observed value (1×10^{-11} mol/L). At intermediate ionic strengths (between 0.01 and 0.05 mol/kg), irreversible colloidal concentration was calculated by interpolating between concentrations of 1×10^{-11} and 6×10^{-8} mol/L.

The stability or tendency of the colloids to remain suspended in the fluid depends on interactions between the surfaces of the colloids, which, in turn are affected by aqueous chemical conditions. Experimental evidence shows that waste form colloids are composed of smectite clay minerals. Consequently, their stability becomes increasingly sensitive to ionic strength as pH drops below about eight. This relationship was captured by a simple linear function that provides a means for the irreversible HLW colloid concentration to be adjusted to reflect solution conditions (Figure 3.5-19).

Potential Concentrations for Corrosion-Product Colloids–Data are not available on the concentration of corrosion product (i.e., iron-(hydr)oxide) colloids that may be generated as a result of corrosion of the waste package materials. Consequently, it was assumed that the initial concentration of these colloids was similar to that of the iron-(hydr)oxide) colloids found adjacent to the iron-rich rock at the Morro de Ferro natural analogue site. This approach is reasonable, because colloid concentrations observed in natural waters represent concentrations developed over a wide range of hydro-geochemical conditions. The stability behavior of corrosion-product colloids, with respect to ionic strength and pH was then estimated. At very low and high pH values, iron-(hydr)oxide colloids are affected by ionic strength, similar to minerals. Thus, high ionic strengths were assumed for destabilization. At intermediate pH values, only relatively low ionic strengths were assumed. The result is the typical "U-shaped" stability curve (Figure 3.5-19). The minimum and maximum mass of corrosion product colloids was 10^{-3} and 1 mg/L, respectively

Potential Colloid Concentrations for Groundwater Colloids–Several workers have studied and compiled the characteristics of colloids in groundwaters from throughout the world (e.g., crystalline and sedimentary rocks; saturated and unsaturated hydrologic regimes). Data on colloid concentration and ionic strength for these groundwaters show a general inverse correlation above an ionic strength of about 0.01 mol/kg and provide an analytical tool for estimating initial groundwater colloid mass concentration. These data were converted to mass or surface area per unit volume, based on the conservative assumption that the colloid populations measured consisted of strongly sorbing smectite clay minerals. Given that assumption, the stability relationship developed for irreversibly attached colloidal material from HLW (described above) was then used to determine the potential groundwater colloid concentration (Figure 3.5-19). The minimum and maximum mass of groundwater colloids was 3×10^{-2} and 3×10^{-6} mg/L, respectively.

Colloid Concentrations for Mobile Colloidal Radionuclide Source Term– The concentrations of reversibly sorbed radionuclides were determined from the potential concentration evaluated above for all three sources of colloids using appropriate sorption partion coefficient (K_d) values from Yucca Mountain-specific sorption data. The total mobile colloidal radionuclide source term was then calculated as the sum of the radionuclide contributions from HLW colloids (reversibly and irreversibly attached), corrosion product colloids, and groundwater colloids (Figure 3.5-19).

3.5.6.2 Implementation in the Total System Performance Assessment

As described in Section 3.5.6.1, the mobile colloidal-radionuclide source term consists of three colloid types: (1) waste form colloids, (2) corrosion-product colloids, and (3) groundwater colloids. For codisposal waste packages, the colloidal source term includes waste form smectites produced from HLW glass, iron-(hydr)oxide colloids produced from degradation of the steel packaging material, and naturally occurring groundwater colloids found in the infiltrating water. Only plutonium and americium are assumed irreversibly attached in waste form colloids and, along with their daughter products, are permanently embedded within the colloid. For CSNF packages, only reversible uptake of radionuclides iron-(hydr)oxide and naturally occurring groundwater colloids occur. The TSPA-SR calculations use in-package and in-drift pH and ionic strength conditions to calculate the generation and stability of waste form colloids and the stability of corrosion-product and groundwater colloids in the waste package and in the invert, respectively.

Consistent with the conceptual model, the following constraints and assumptions were used to implement the TSPA-SR model:

- Colloids with embedded radionuclides (i.e., irreversibly attached) are created as glass waste degrades. They contain a known initial mass of the parent radionuclide but no initial mass of the daughter products.

- Colloids with reversibly sorbed radionuclides are not created until the maximum concentration of irreversible colloids is satisfied in a given time step.

- Irreversibly attached radionuclides are partitioned onto colloids according to their isotopic mass fraction at the time of the formation of the colloids.

- Colloids with irreversibly attached radionuclides also act as substrates for reversible sorption.

- Thorium, protactinium, plutonium, and americium (and their daughters) can be transported as colloids.

- Daughters of parent radionuclides in irreversible colloids are bound within the colloids for the duration of their travel through the system. For irreversibly bound ^{242}Pu, ^{240}Pu, and ^{239}Pu, the combined dose of the daughters and granddaughters within the colloid is several orders of magnitude less than the dose of the parent during the travel time within the natural barriers, and decay to daughter products is not implemented. ^{243}Am is decayed to ^{239}Pu, but the daughters of ^{239}Pu are neglected because of their low impact on dose. Irreversibly bound ^{241}Am and ^{238}Pu are decayed to irreversibly bound ^{237}Np and ^{234}U daughters, respectively.

- CSNF waste does not produce colloids. However, CSNF radionuclides can reversibly attach to naturally occurring smectites and iron-(hydr)oxide colloids.

- There are naturally occurring smectite colloids that will be sites for reversibly sorbing radionuclides in the water entering the CSNF and codisposal waste packages.

- Diffusion of both irreversible and reversible colloids is permitted and the diffusivity is conservatively set at only 100 times less than the diffusivity of the aqueous phase.

- For far-field transport, the K_c equilibrium model is used for the transport of radionuclides reversibly bound to colloids as described in Section 3.8.2.1.

For a detailed discussion of TSPA-SR treatment of colloids, refer to the logic flow charts in *Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary* (CRWMS M&O 2000 [148214]).

Irreversible-Colloid Mass in the Waste Form Cell–Accounting for the concentration of irreversibly attached radionuclides, temporally and spatially, in TSPA-SR is challenging because of the need to track masses of colloidal and dissolved radionuclides. In the TSPA-SR model, the mass of irreversibly bound radionuclides is tracked through the use of surrogate species that represent the radioisotope mass embedded in waste form colloids. The masses of the surrogate species, or irreversibly bound radioisotopes, are proportional to the amount of waste form exposed using the high-level radioactive waste glass dissolution rate. To start the TSPA-SR calculations, an initial inventory of radioisotopes must be provided. For the surrogate species, an arbitrarily large value is used, so that the irreversibly bound colloid species cannot become inventory limited. In other words, irreversible colloids are allowed to form and be released as long as radioisotopes are present in the high-level radioactive waste glass waste form cell.

The mass of a specific irreversibly bound radioisotope is calculated in each time step by considering colloid mass flux (i.e., mass of colloids per year), time, the concentration of irreversibly attached radioisotope per solution volume, and colloid mass in that volume. That mass is subtracted from the total radioisotope mass in the waste form cell. Subtracting the irreversibly bound mass first ensures that radioisotopes are assigned to irreversible colloids first, which is conservative with respect to radionuclide transport. The masses of all forms of radioisotopes released, including irreversibly attached ones, are tracked to ensure that the available inventory mass is not exceeded.

Irreversible Colloid Mass Flux from the Waste Form Cell–The irreversible colloid-mass flux from the waste form cell is estimated through two sequential steps. First, the generation of irreversible colloids is calculated, based on consideration of ionic strength. Second, the stability of the colloids is calculated, based on both ionic strength and pH. Because of this sequence, the first step essentially provides the upper bound on concentrations of irreversible colloids, and the flux of the irreversible colloid species is defined by the stability of the colloids.

Calculation of Plutonium and Americium Concentrations Irreversibly Attached to Colloids–The total concentration of plutonium bound irreversibly in waste form colloids is partitioned into the plutonium isotopes based on the mole fraction in the source term for codisposal waste packages. The plutonium mass fraction in the source term is calculated by dividing the mass of each plutonium species exposed by the total mass of plutonium released for each time step. For americium, the concentration bound irreversibly in waste form colloids is determined as a function of the mass fraction of americium to plutonium in the source term. By using the total concentration of plutonium bound irreversibly, and the ratio of americium to plutonium in the source term, the total concentration of americium bound irreversibly is calculated.

Reversible Colloid Mass Flux from the Waste Form Cell–Each combination of colloid type (waste form, corrosion product, and groundwater) and critical radionuclide has a specified K_d value representing reversibly sorbed radionuclide concentrations. To calculate reversible radionuclide concentration in each waste form environment, colloid masses are determined based on ionic strength and, in all cases but groundwater colloids, pH. Masses are then used with dissolved radionuclide concentrations and K_d values to determine a concentration associated with colloids. Masses are determined by considering mass flux for each colloid type and the volume of water in the waste form environment.

3.5.6.3 Treatment of Uncertainty and Variability

The colloid model abstraction is based on laboratory results from waste form corrosion testing and testing of adsorption and desorption properties of plutonium and americium on clay and iron-(hydr)oxide colloids. The development of the conceptual model and implementation requires consideration of colloid generation, colloid-radionuclide interaction, colloid stability behavior, and, to some extent, colloid transport and retardation behavior. Information used for groundwater colloids, waste form colloids, and corrosion product colloids was obtained primarily from Yucca Mountain-specific studies.

Uncertainty was explicitly represented in statistical distribution for the K_d parameter values, where an uncertainty band of plus or minus one order of magnitude was assigned to each value. Because bounding values or estimates were used for all other parameters, the greatest source of uncertainty stems from conceptual models. Mass concentrations of groundwater colloids are bounding. Mass concentrations of corrosion product colloids are linked to concentrations of groundwater colloids on the basis of similar behaviors (as mineral colloids) in natural waters, and are, therefore, bounding. Concentration of irreversibly attached plutonium waste form colloids generated from HLW are bounding, based on experimental results. The concentration of irreversibly attached americium waste form colloids was assumed to be proportional to that of plutonium. Justification for this assumption is lacking; however, the fact that americium is highly reversibly sorbed because of the high K_d values used at least permits a large percentage of the americium inventory to escape as colloids. Furthermore, the proportion of americium released as colloids is greater than for plutonium since the K_d value for plutonium is less than the K_d value for americium.

The primary uncertainty is associated with the limited information available on formation of colloids from degradation of N-Reactor fuel, and its potential contribution to the potential repository performance. A relevant research program is ongoing. For now, however, it is assumed in the model that, due to the small quantity of N-reactor fuel, any colloids generated from degradation of DSNF fuel will have little or no effect on the potential repository performance. Corrosion product colloids, as well as groundwater colloids, are associated with all waste types, and provide a colloidal transport mechanism.

Another, but less significant, uncertainty is the treatment of colloids produced as a result of CSNF degradation. Results from the Argonne National Laboratory tests on degradation of CSNF (mentioned in Section 3.5.5.3) indicate that very little colloidal material was produced. Colloids formed were smectites, some with apparently adsorbed plutonium and uranium silicates. However, examinations showed that no embedded radionuclide phases were in the few clay colloids that were produced during the degradation testing. If any of the colloids contain embedded (irreversibly attached) radionuclides, the consequences for waste package releases would be minimal, since it is assumed that all colloid-associated radionuclides leave a failed waste package. As with DSNF, corrosion product colloids, as well as groundwater colloids, are associated with all waste types, and provide a colloidal transport mechanism. This is particularly true for CSNF, in that groundwater colloids are relatively abundant and smectite has mineralogy identical to the few observed CSNF colloids.

3.5.6.4 Results and Interpretation

The radionuclides ^{99}Tc, ^{129}I, ^{237}Np, and ^{239}Pu are important contributors to the dose as discussed in Section 4. Even though colloidal americium was modeled with a conservatively high sorption coefficient that should have made americium readily available for transport as a reversible colloid, the concentration of americium (either aqueous or colloidal) was not an important contributor to the dose. Of the four important radionuclides, ^{239}Pu is transported both in the dissolved phase and attached to colloids. However, the contribution of colloids to the overall release of ^{239}Pu is not important. The dissolved phase completely dominates the overall release (Figure 3.5-20a). As implemented in the TSPA-SR, both types of colloids, reversible and irreversible, contribute equally to the dose although this contribution is insignificant overall. The source of the reversible colloids is primarily from the waste itself; natural groundwater colloids and rust colloids are only somewhat important (Figure 3.5-20b). As noted in the previous section, several assumptions in the colloidal model introduce uncertainty (e.g., assumptions on DSNF colloids and colloids containing americium); however, because of the small contribution of colloids to the overall dose releases (Figure 3.5-20a), and because of the small contribution of codisposal packages to the total dose as discussed in Section 5.2.4, changes to these assumptions to improve the model will have only minimal impact on the overall dose.

3.6 ENGINEERED BARRIER SYSTEM TRANSPORT

Radionuclide transport out of the waste form and waste package, through the invert, and into the UZ is dependent on a complex series of events in the potential repository.

After the waste packages are emplaced, radioactive decay of the waste will heat the drifts. This heating process may evaporate some (or all) of the groundwater near the drifts, thereby perturbing the natural flow pattern for percolation of water through the mountain. As the drifts cool and the natural flow pattern is reestablished, some of the water percolating through the mountain may seep into the drifts and contact some of the drip shields. Over time, the drip shields and waste packages are expected to degrade. Once this occurs, water can contact the waste form, resulting in the mobilization and transport of radionuclides through the EBS.

The primary transport medium through the EBS is (liquid) water. This water may be flowing or dripping slowly through the EBS. Alternately, this water may form a continuous film of stationary liquid. Either condition must be present for mobilization of radionuclides in the waste form and transport of radionuclides through the invert and into the UZ. A dry waste package will not release any radionuclides.

Gaseous transport of volatile radionuclides has been screened out of the nominal scenario because of low consequence (see FEP 3.2.10.00.00 in CRWMS M&O 2000 [150806]). This FEP includes consideration of radiotoxic and chemotoxic species in the air as gas, vapor, particulates, or an aerosol. The radionuclide with the greatest potential for gaseous release is ^{14}C. Bounding estimates of the potential dose of ^{14}C indicate that the maximum release of ^{14}C will be at least 5 orders of magnitude below the anticipated regulatory dose limit. It follows that gaseous transport is not included in the EBS flow and transport abstraction for the TSPA.

As an aside, atmospheric transport due to volcanic ashfall is included in the TSPA for the igneous intrusion scenario (CRWMS M&O 2000 [150806]). This scenario is not relevant to EBS flow and transport because the EBS is destroyed by the igneous intrusion process (see FEP 1.2.04.07.00, Ashfall in CRWMS M&O 2000 [150806]).

The primary components of the EBS are a drip shield, a waste package on an emplacement pallet, and an invert. The invert is a metal cage that provides structural support for the waste package and emplacement pallet; it is filled with crushed tuff. Figure 3.6-1 presents a typical cross-section of an emplacement drift with the major components of the EBS.

Each component of the EBS is designed to prevent (or delay) the mobilization and release of radionuclides into the geologic environment. The drip shield is designed to redirect seepage that flows into the drift away from the waste package. The waste package is fabricated from a double shell of corrosion-resistant material and corrosion-allowance material to minimize and delay waste package failures in the potential repository environment. The emplacement pallet holds the waste package above any liquid that might pool on the invert and allows the waste package to shed water. The invert can act as a barrier to diffusive transport in liquids, if the liquid saturation in the crushed tuff is low.

Once a drip shield is breached, water may contact the waste package. Once a waste package is breached, water may enter the package as water vapor or as drips. This water will come into contact with the metal cladding of commercial spent nuclear fuel rods or with the stainless steel canister surrounding vitrified high-level waste. If the metal cladding that encases the spent nuclear fuel pellets is also breached, radionuclides may dissolve in the water and be transported out of the waste package. Similarly, if the stainless steel canister surrounding the vitrified waste form is breached, the glass and its associated radionuclides may dissolve and be mobilized for transport out of the waste package. Figure 3.6-2 illustrates this transport process for patches formed by general corrosion of the drip shield and waste package. As shown in this figure, the patches provide a path for dripping water, also called an advective flux, to enter the top of the waste package, contact the waste form, mobilize radionuclides, and carry these radionuclides out the bottom of the waste package.

The concentration of each mobilized radionuclide cannot exceed the radionuclide solubility limit, unless colloids are present. Colloids are small particulates, with a typical size range of 10^{-3} to 10^{-6} mm, that can physically or chemically bind with ions in an aqueous environment. They often occur naturally in the geologic environment because clay minerals and some geochemical weathering products of rocks are of colloidal size and can persist in aqueous solution for long periods of time. Colloids are important for transport because they can increase the mobilized concentration of radionuclides above the solubility limit. Colloids can also increase the transport velocity of radionuclides, because they have been observed to move through pore spaces more quickly than dissolved radionuclides. This effect is neglected in the EBS because the typical size scale of the invert, about 1 m deep in cross section, is much less than the typical size scale of the UZ, on the order of hundreds of meters. In this situation, the advective travel time through the EBS will be negligible in comparison to the advective travel time through the UZ, so the effective velocity of colloids in the EBS can be neglected for performance assessment of the total system.

Radionuclides that are mobilized as dissolved or colloidal species may be transported by advection or by diffusion. Advective transport occurs when dissolved or suspended radionuclides are carried along with a flowing liquid. Advective transport is anticipated to be the dominant transport mechanism through patches formed by general corrosion (see Figure 3.6-2). Diffusive transport occurs when dissolved or suspended radionuclides move from regions of higher concentration to regions of lower concentration. Diffusion is anticipated to be the main transport mechanism through stress corrosion cracks, as illustrated in Figure 3.6-3. Diffusion will be the dominant mechanism through a crack because the small size and associated capillary forces within a crack prevent significant advective fluxes from passing through. Even though advection is excluded, each crack is assumed to retain moisture in the form of a thin, liquid film that provides a continuous pathway for diffusive transport at all times.

Once outside the package, radionuclides will be transported through the invert by diffusion (if water is not flowing through the invert), or by diffusion and advection (if an appreciable amount of water is dripping through the invert).

3.6.1 Construction of the Conceptual Model

A number of factors will affect the mobilization and transport of radionuclides through the EBS. These factors are:

- Performance of the drip shield
- Performance of the waste package
- Protection provided by cladding
- Waste form dissolution rates
- Entry and movement of water through the waste package
- Solubility limit for each radionuclide
- Transport of radionuclides through and out of the waste package
- Transport of radionuclides through the invert
- Transport of radionuclides via colloids.

Given these multiple factors, the conceptual model for EBS transport requires input and information from many elements of the TSPA. Figure 3.6-4 illustrates this transfer of information for the two parts of the conceptual model: the EBS flow abstraction and the EBS transport abstraction.

The EBS flow abstraction has three major inputs. The first input is the seepage flux abstraction that defines the fluid flux into the EBS as a function of time, location within the potential repository, and the climate state. The second input is the drip shield and waste package degradation models that define the type, number, and timing of breaches in these components. The third input is the abstraction of the thermal-hydrologic response of the near-field environment that defines the time-dependent temperature, RH, and evaporative fluxes in the EBS.

The EBS transport abstraction has two major inputs. The first input is the output from the EBS flow abstraction that defines the fluid fluxes through the waste package and invert as a function of the time-dependent conditions in the EBS. The second input is the abstractions for waste form dissolution rate, radionuclide solubility limits, and colloidal concentrations that are required to define the mobilized concentration of radionuclides for advective and diffusive transport.

The conceptual models for the EBS flow abstraction and the EBS transport abstraction are described in greater detail in the next two sections.

3.6.1.1 Engineered Barrier System Flow Conceptual Model

The conceptual model for the EBS flow abstraction is illustrated in Figure 3.6-5. This figure includes the key elements of the abstraction, the inputs to and outputs from the abstraction, and identifies the experimental data that provide a basis for confidence in the abstraction.

The key elements of the conceptual model for EBS flow abstraction are:

- **One-dimensional, quasi-steady flow**–Although flow through the EBS may be complex and multi-dimensional, it is abstracted to a one-dimensional network of flow pathways.

The flow system is also assumed to be quasi-steady. This means that fluid immediately flows through the system and does not accumulate within the EBS.

- **Flow-through model for the waste package**–This assumption specifies that fluid does not accumulate in the waste package. The flow-through model has been found to be conservative relative to an alternate conceptual model (called the bathtub model) that allows fluid to accumulate in the waste package. (CRWMS M&O 2000 [129284], Section 6.6). While the conceptual model for waste form degradation and mobilization (see Section 3.5) specifically assumes a saturated, well-mixed environment in the waste package, the EBS flow abstraction is based on the flow-through model in order to maximize immediate release from the waste package into the EBS.

- **Effects of drip shield and waste package degradation**–The type, number, and timing of breaches in the drip shield and waste package are predicted by the WAPDEG code. This information is used by the EBS flow abstraction to define the time-dependent fluxes that flow through (or are diverted around) the drip shield and the waste package.

- **Thermal and mechanical response of the drip shield**–A fluid pathway can be created if drip shields separate in response to rock fall, seismic events, or thermal expansion. Detailed analyses of these processes have shown that separations will not occur, so these effects have been screened out of the conceptual model for the EBS for the TSPA-SR (CRWMS M&O 2000 [146538]).

Inputs to the flow abstraction are primarily from other elements of the TSPA (see Figure 3.6-5). These inputs include:

- The flux of fluid into the EBS, as defined by the seepage flow abstraction (CRWMS M&O 2000 [142004])

- The temperature, RH, saturation, and evaporative flux from the invert, as defined by the abstraction of thermal hydraulic calculations (CRWMS M&O 2000 [152204])

- The timing, size, number, and location (upper or lower surface) of breaches in the drip shield and waste package, as defined by the results from WAPDEG analyses (CRWMS M&O 2000 [146427]).

Outputs from the flow abstraction include the time-dependent fluxes through the drip shield, waste package, and invert. These fluxes are used by the EBS transport abstraction to determine the mass of radionuclides released to the UZ.

3.6.1.2 Engineered Barrier System Transport Conceptual Model

The conceptual model for the EBS transport abstraction is illustrated in Figure 3.6-6. This figure includes the key elements of the abstraction, the inputs to (and outputs from) the abstraction, and identifies the experimental data that provide a basis for confidence in the abstraction.

Radionuclides mobilized as dissolved or colloidal species may be transported by advection or diffusion. Diffusion will be the dominant transport process through SCC because the advective flux through cracks is expected to be negligible (CRWMS M&O 2000 [129284], Section 6.3.2). Advection is expected to be the dominant transport process through any patches that form in the drip shield and waste package when an appreciable amount of water is dripping through the waste package and invert.

The EBS transport abstraction is general in nature, allowing simulation of EBS transport in response to the time-dependent breaching of the drip shield and waste package. These breaches may take the form of patches, pits or stress corrosion cracks, as explained below. The specific corrosion process that generates these breaches will depend on the near-field physical and chemical environment and the materials in the drip shield and waste package. The key elements of the EBS transport abstraction are:

- **One-dimensional, quasi-steady flux through the EBS**–This flux is determined by the EBS flow abstraction described in Section 3.6.1.1.

- **Advective transport through breaches in the waste package**–Breaches in the waste package may be created by various corrosion mechanisms. Patches can be created by general corrosion. Pits can be created by localized corrosion, also called pitting or crevice corrosion, induced by variations in the electrochemical potential on a microscale over small regions. Both patches and pits are conceptualized to have a large enough cross-sectional area that they will provide a pathway for advective flow and transport through the waste package. Radionuclides will be mobilized once an advective flux exists in the waste package and invert.

- **Retardation and concentration of radionuclides**–The EBS transport abstraction maximizes release to the UZ by conservatively assuming no retardation of dissolved species in the waste package or invert. In addition, forward and lateral dispersion is ignored because of the small thickness of the invert. The dissolved concentration of a radionuclide can be increased above the solubility limit if stable colloids physically or chemically bond with the radionuclide.

- **Diffusive transport through stress corrosion cracks, patches, and pits**–Radionuclides can be transported by diffusion through any breach in the waste package. In theory, the waste form may dry out due to heat generation for a few thousand years after emplacement, and diffusive transport cannot occur if no liquid is present. However, diffusion is conservatively assumed to always occur once a breach forms in the waste package independent of the predicted in-package condition of the waste form.

The inputs to the transport abstraction are primarily from other elements of the TSPA:

- The (advective) flux of fluid through the waste package and invert, as defined by the EBS flow abstraction

- The temperature and saturation of the invert, as defined by the abstraction of thermal hydrologic calculations (CRWMS M&O 2000 [152204])

- The source term for radionuclide mobilization, as defined by abstractions for (1) the dissolution rate of SNFs and vitrified waste forms, (2) the solubility limits of radionuclides, and (3) the density of colloidal particles and their partition coefficients for the radionuclides in the system

- The time-dependent diffusion coefficient for the invert, which is a function of the self-diffusivity of water, D_0, and the porosity, ϕ, liquid saturation, s, and temperature, T, of the invert. The functional dependence of the diffusion coefficient, D, on these parameters is given by

$$D = D_0 s^{1.849} \phi^{1.3} f(T) \qquad \text{(Eq. 3.6-1)}$$

where $f(T)$ is a function of temperature based on the Nernst-Einstein equation.

Output from the EBS transport abstraction is the mass of radionuclides released to the UZ.

3.6.2 Implementation in the Total System Performance Assessment

3.6.2.1 Engineered Barrier System Flow Abstraction

The source of inflow to the EBS is the seepage flux into the drift that results from the downward infiltration of fluid through the existing fracture system at Yucca Mountain. The seepage flux is conceptualized to flow from discrete fractures above the roof of the drift, falling vertically downward onto the drip shield, the invert, and the waste package, if the drip shield has been breached. The seepage flux is represented as an abstraction for the EBS flow and transport model (CRWMS M&O 2000 [142004]).

The seepage flows through the EBS along eight pathways, as shown in Figure 3.6-7. The pathways are:

1. **Seepage flux entering the drift**–This is the fluid flow into the EBS.

2. **Flow through the drip shield**–Fluid flux through the drip shield begins once patches form due to general corrosion. The number of patches through the drip shield is calculated independently of the EBS flow abstraction by the WAPDEG code (CRWMS M&O 1998 [145618]). The nominal size of a patch is fixed for the WAPDEG calculations (CRWMS M&O 2000 [146427]). It is currently defined to be 7.21×10^4 mm^2, equivalent to a square 10.6 inches on a size (CRWMS M&O 2000 [146427]).

   The fluid flux through any patches in the drip shield is proportional to the seepage flux entering the drift, times the ratio of the axial length of all patches in the drip shield to the total axial length of the drip shield. The rationale and conservatism of this algorithm, called a flux splitting algorithm, is discussed later in this section. The algorithm is specifically for patches because pitting of the titanium drip shield is not expected to occur in the near-field geochemical environment (CRWMS M&O 2000 [144229]).

3. **Flow diversion around the drip shield**–The portion of the flux that does not flow through the drip shield is assumed to bypass the invert and flow directly into the UZ. This approach is consistent with a quasi-static flow because the sum of the fluid volume entering the drip shield (Pathway 2) and the fluid volume diverted around the drip shield (Pathway 3) must equal the fluid volume entering the EBS (Pathway 1) for a steady state system. Diversion directly to the UZ is also reasonable because diverted flow does not contact the waste form and is not contaminated with radionuclides. It is, therefore, ignored by the EBS transport abstraction.

4. **Flow through the waste package**–The fluid flow through the waste package is based on the presence of patches due to general corrosion. The number of patches through the waste package is calculated independently of the EBS flow abstraction by the WAPDEG code. The nominal size of a patch is fixed for the WAPDEG calculations. It is currently defined to be $2.346 \times 10^4 mm^2$, equivalent to a square 6 inches on a size (CRWMS M&O 2000 [146427]). The area of each stress corrosion crack, $4.08 \times 10^{-6} m^2$ or $0.0063 in^2$, is estimated from process level calculations of the residual stress in the welded lids of the waste package. This area corresponds to a hole with an elliptical cross section that is 1.6 inches long by 0.005 inches wide (CRWMS M&O 2000 [129284], Section 6.3.1.2.1).

   The fluid flux through any patches in the waste package is proportional to the seepage flux falling on the waste package, times the ratio of the axial length of all patches in the waste package to the total axial length of the waste package. The seepage flux falling on the waste package is equal to the fluid flux through the drip shield. The rationale for and conservatism of the flux splitting algorithm is discussed later in this section. The algorithm is defined for patches because pitting of the Alloy-22 outer shell of the waste package is not expected to occur in the near-field geochemical environment and because the advective flux through stress corrosion cracks will be negligible (CRWMS M&O 2000 [144229]).

5. **Flow diversion around the waste package**–The portion of the flux that does not flow into the waste package is assumed to bypass the waste form and flow directly to the invert. This approach is consistent with no accumulation of fluid for a quasi-static flow.

6. **Evaporation from the invert**–The magnitude of the evaporative flux from the invert is based on the thermal hydraulic abstraction (CRWMS M&O 2000 [152204]). If the drip shield is cooler than the invert, then all the evaporative flux is assumed to drip on the waste package. If the drip shield is hotter than the invert, then there is no dripping on the waste package from the evaporative flux. The magnitude of the evaporative flux and the temperatures of the drip shield and invert are pre-calculated and abstracted to provide runtime input data for the EBS model in the TSPA-SR. The rationale for this approach is explained in EBS Radionuclide Transport Abstraction (CRWMS M&O 2000 [129284], Section 6.3.3).

7. **Flow from the waste package, to the invert**–All flux from the waste package flows to the invert, independent of breach location on the waste package. The presence of the emplacement pallet is ignored, and the waste package is assumed to be lying on the invert so that a continuous liquid pathway for diffusive transport exists at all times.

8. **Flow through the invert into the UZ**–All fluid and mass flux into the invert is immediately released into the UZ, consistent with the quasi-static assumption for flow through the EBS.

These pathways are time dependent because breaches of the drip shield and waste package will vary with time and local conditions in the potential repository. For example, at very early times there may be no penetrations through the drip shield, so fluid can reach the waste package only if pathway six (evaporation from the invert and condensation on the drip shield) is active. Once patches have formed on the drip shield, fluid can enter the waste package if any patches or stress corrosion cracks exist in the waste package.

The most important element of the EBS flow abstraction is the flow splitting algorithm that determines the fluid volume that flows through the drip shield or waste package and the remainder that flows around the drip shield or waste package. This algorithm assumes that the fluid flux through any patches or pits in the drip shield or waste package is proportional to the ratio of the total length of all patches or pits in the axial direction to the total axial length of the drip shield or waste package. This algorithm is equivalent to assuming that a patch or pit will collect all fluid that drips or flows onto the drip shield or waste package at the same axial location as the patches or pits. This algorithm is conservative because drips on the right-hand side of a drip shield or waste package will contribute to the flow through a patch/pit on the left-hand side. This is not physically possible because the droplets cannot flow uphill, against the direction of gravity. An additional conservatism of this model is that there will be a nonzero flux through the waste package in situations where flow is impossible, such as with a single patch or pit or with patches or pits on only the upper half of the drip shield or waste package.

The performance assessment model for flow through the EBS includes two mixing cells, one for the waste package and waste form and a second for the invert. A mixing cell is a volume of fluid with well-mixed, homogeneous conditions throughout. The two mixing cells are conceptualized as having a cylindrical, concentric, one-dimensional geometry for volume calculations. The diameter of the first cell is equal to the diameter of the appropriate type of waste package. The second cell (invert) wraps around the lower half of the waste package and has a thickness of 0.606 m (about 2 feet), equal to the maximum thickness of the invert directly beneath the waste package. This value is appropriate because mass transport out of the waste package is primarily vertically downward beneath the package and centered over the thickest part of the invert.

The waste package mixing cell represents the source term for the TSPA-SR. Source term abstractions for radionuclide solubility, dissolution rate, cladding response, and inventory by waste package type are defined in Section 3.5. The source term represents input data or boundary conditions for the EBS transport abstraction, which is discussed in the next section.

The final output from the EBS flow abstraction is the flux of fluid along the eight pathways of the EBS. Of these pathways, only pathways seven and eight carry contaminants through the EBS because they are downstream from the waste form. Other flows, such as the diversionary flows from the drip shield and waste package (pathways three and five), do not carry mobilized radionuclides and are not required by the EBS transport abstraction.

3.6.2.2 Engineered Barrier System Transport Abstraction

The waste form is the source of all radionuclides considered for the EBS. Radionuclides can be transported downward through the invert and into the UZ, as shown in Figure 3.6-8. Transport can occur by advection when there is a fluid flux through the waste package and invert. Transport can also occur by diffusion through stress corrosion cracks and through the invert to the UZ. The EBS transport abstraction conservatively assumes that diffusion can occur once stress corrosion cracks form, regardless of whether conditions are appropriate for a continuous liquid pathway to exist.

Colloid-facilitated transport of radionuclides is included as a transport mechanism in this abstraction. Radionuclides are transported from the waste package as either dissolved species or bound in and/or attached to colloids. There are three types of colloids in the EBS: (1) waste form colloids, (2) corrosion product (iron oxy-hydroxide) colloids, and (3) groundwater colloids. The waste form colloids may have irreversibly attached (embedded) or reversibly attached (sorbed) radionuclides. The corrosion and groundwater colloids may have reversibly attached radionuclides.

The diffusion coefficient in the invert is evaluated in three steps, (1) determine the diffusion coefficient of each radionuclide in water (often called the free water diffusion coefficient), (2) correct this value for the presence of the invert, and (3) correct this value for the presence of a colloid. This second step is necessary because the invert is a porous, partly saturated medium that provides more resistance to diffusion than an aqueous solution. The third step is necessary for colloids because they are usually much larger than a dissolved anion or cation and therefore diffuse much more slowly.

Each radionuclide complex could be assigned a unique value for its free water diffusion coefficient in the first step. The complexity of the time-dependent chemistry in the waste package makes this a challenging task, so a simpler approach has been incorporated into the TSPA-SR. The value for the free-water diffusion coefficient for all radionuclides is set equal to the self-diffusivity of water at 25°C. This is a reasonable and conservative approximation because the self-diffusivity of water has been shown to be a bounding value for all radionuclides of interest in the TSPA-SR. (CRWMS M&O 2000 [129284], Section 6.4.1).

The value for the self-diffusivity of water must then be corrected for invert porosity and for the time-dependent invert saturation and invert temperature. The effects of porosity and saturation are included using a variant of Archie's law for a partly saturated porous medium. The effect of temperature is included using the Nernst-Einstein formulation. The functional dependence of these corrections is given by

$$D = D_0 s^{1.849} \phi^{1.3} f(T) \qquad \text{(Eq. 3.6-2)}$$

where D is the diffusion coefficient, D_0 is the self-diffusivity of water at 25°C, s is the liquid saturation in the invert, ϕ is the porosity of the invert, and $f(T)$ is the correction for invert temperature, T, based on the Nernst-Einstein equation. (CRWMS M&O 2000 [129284], Section 6.4.1).

Finally, the diffusion coefficient for radionuclides bound to colloids is given by the diffusion coefficient for a dissolved radionuclide divided by 100. That is,

$$D_{colloid} = D/100 = (0.01)D_0 s^{1.849}\phi^{1.3} f(T) \quad \text{(Eq. 3.6-3)}$$

The rationale for the factor of 100 reduction in diffusion coefficient for colloids is discussed in Section 3.5.6.2 of this document.

Diffusive transport calculations through the invert also require boundary conditions on the top and bottom of the invert. The boundary condition on the top of the invert is simply the radionuclide concentration within the waste package. The boundary condition on the bottom of the invert, at the boundary with the UZ, is a zero concentration (swept away) boundary condition. The zero concentration boundary condition is implemented by defining a flow cell with a small volume of water but a very high advective outflow, effectively sweeping all radionuclides away from the lower boundary. This is a reasonable approximation when advective fluxes are large because advection will tend to sweep radionuclides away from the invert, diluting the local concentrations below the invert. This can be a conservative assumption if the dominant transport mechanism is diffusion and the advective flux beneath the waste package and invert is low.

Products from the corrosion of the waste package and spent fuels have the potential to strongly sorb actinides. Sorption on corrosion products will be beneficial to performance because this process can retain radionuclides in the EBS and delay their release to the UZ. However, the effects of retardation are conservatively ignored in the EBS transport abstraction. Table 3.6-1 summarizes the modes and parameters for the transport abstraction.

Table 3.6-1. Summary of Transport Modes and Parameters for the EBS Transport Pathways

Transport Pathway	Transport Modes	Transport Parameters and Data Sources
7. Waste package to invert	Diffusion through stress corrosion cracks (no advective transport occurs through stress corrosion cracks) Diffusion and advection through patches Diffusion and advection through pits (no pits are expected in Alloy-22 (CRWMS M&O 2000 [144229])	Fluid flux for advection, F_7, equals the flux through the waste package, F_4; Diffusive area for each stress corrosion cracks is given by 4.08×10^{-6} m^2 Diffusive area for each patch is 2.346×10^4 (mm)^2 Diffusive length in waste package is 135 mm to 185 mm, depending on waste package type Free-water diffusion coefficient for all radionuclides is 2.299×10^{-5} cm^2/s at 25°C - Corrected for porosity, saturation and temperature (CRWMS M&O 2000 [129284], Section 6.4.1) - Reduced by a factor of 100 if radionuclide is attached to a colloid

Table 3.6-1. Summary of Transport Modes and Parameters for the EBS Transport Pathways (Continued)

Transport Pathway	Transport Modes	Transport Parameters and Data Sources
8. Invert to UZ	Diffusion and advection through the invert Flow cross-sectional areas in the invert are given by $A_{Invert} = \pi(R_{WP})L_{WP}$ $A_{UZ} = \pi(R_{WP} + \Delta r_{invert})L_{WP}$ where R_{WP} is radius of waste package, L_{WP} is length of waste package, and Δr_{invert} is the thickness of the invert (0.606 m)	Fluid flux for advection, F_8, equals the flux through the invert, F_7; Diffusive length = 0.606 m (maximum thickness of invert); Free-water diffusion coefficient for all radionuclides is 2.299×10^{-5} cm^2/s at 25°C - Corrected for porosity, saturation and temperature (CRWMS M&O 2000 [129284], Section 6.4.1) - Reduced by a factor of 100 if radionuclide is attached to a colloid The flow cross-sectional areas, A_{invert} and A_{UZ}, assume a cylindrical geometry, corresponding to the lower half of the waste package lying in contact with the invert. A_{invert} is one-half of the surface area of the waste package and A_{UZ} is the corresponding surface area at a radius equal to the radius of the waste package plus the maximum thickness of the invert. The invert diffusion calculation uses radionuclide concentrations in the waste package as the boundary condition at the top of the invert and a zero concentration (swept away) boundary condition on the bottom of the invert, at the interface with the UZ.

3.6.3 Results and Interpretation

Results with the EBS flow and transport abstraction are presented in Section 5, Sensitivity Analyses. Rather than repeat this material, this section presents a discussion of the conservatisms in the abstraction.

3.6.3.1 Applicability and Conservatisms

The EBS flow and transport abstractions for the TSPA-SR are based on a reasonable approach that bounds the response of the EBS. The flow abstraction is based on typical flow processes, such as the advective flow of liquid water through the EBS and the potential for evaporation from the invert and condensation of water vapor on the underside of the drip shield. The transport abstraction is based on diffusive and advective transport of radionuclides dissolved in liquid water and bound to colloids. The use of reasonable bounds is appropriate because of potentially large uncertainties in the response of a very complex, engineered system over long periods of time. The EBS flow and transport abstractions are valid and appropriate for their intended use because they are designed to represent fundamental flow and transport processes in a bounding or conservative framework. Following are the noteworthy conservatisms in these abstractions:

- **Seepage through the drip shield is assumed to always falls on a waste package**–The current potential repository design has a small axial gap between adjacent waste packages. It is possible that the seepage through the drip shield will fall directly to the invert, avoiding the waste package entirely. This possibility is conservatively ignored. This is a minor conservatism when the spacing between adjacent waste packages is small compared to the length of each waste package.

- **Seepage is assumed to wet the drip shield and waste package uniformly**–The pathways for seepage into the drifts are fractures or fracture sets. As a result, seepage will vary spatially and temporally over the approximately 10,000 waste packages in the potential repository. Therefore, the response of groups of waste packages is represented as averages for performance assessment. In addition, it is assumed that any breach is located so that it will collect all fluid that drips onto the drip shield or waste package at the same axial location as the breach. This assumption conservatively ignores the fact that fluid dripping onto the lower portion of the drip shield or waste package will not flow through a breach high on the drip shield or waste package. It also conservatively ignores the fact that seepage on the left half of a drip shield or waste package cannot flow through a breach on the right half. The assumption for breach location is thereby conservative by approximately a factor of two for the calculation of fluid flows into the drip shield and waste package.

- **Diffusion is maximized because diffusive transport is always possible through stress corrosion cracks and because the waste package is in contact with the invert**–The waste form is assumed to be covered with a thin film of liquid that supports diffusive transport at all times. In addition, the waste package is assumed to be in contact with the invert, providing a continuous liquid pathway for diffusion. In this situation, radionuclides will be released by diffusion through a stress corrosion crack, even when the drip shield is intact, and there is no advective flux into the waste package. This assumption may be very conservative for several reasons. First, the beneficial effects from the partly saturated and degraded waste form on in-package transport and in-package sorption of radionuclides have been ignored. Second, the beneficial effects from the potential capillary and diffusive barrier formed by corrosion products between the two lids of the waste package have been ignored. Finally, a continuous liquid pathway cannot exist when the pallet is intact and there is no advective flow. Ongoing work for the TSPA-LA may quantify these effects in the future.

- **Release of radionuclides through advective transport is assumed to be independent of the location of breaches on the waste package**–Advective transport out of the waste package is based on a flow-through model that is independent of the location of penetrations through the drip shield or waste package. This means that advective transport will occur, even if a waste package has only one penetration or has one or more penetrations on its upper surface and none on its lower surface.

  This assumption results in radionuclide release before patches exist in both the upper and lower halves of the waste package. In effect, the model is ignoring the delay until the patch geometry supports flow through the waste package. The magnitude of this delay is difficult to estimate because WAPDEG does not conveniently output the patch history for a single waste package and because the delay is a complex function of 5 stochastic parameters.

A preliminary estimate of the maximum delay, based on sampling the stochastic for the corrosion rate of Alloy-22 for 1,000 patches, indicates a value of approximately 5,000 years. This delay can be decreased by a factor of 1 to 5 for the combined effects of aging and of microbially-induced corrosion on the general corrosion rate. This delay may be further increased or decreased by the stochastic parameters for patch-to-patch variability and package-to-package variability. These factors do not change the conservatism of the flow-through model for a single patch, but do make it difficult to estimate the magnitude of this conservatism because of the complex statistical behavior of waste package degradation.

- **Evaporation within and on the waste package is ignored**–Diffusive transport will cease if the heat from the waste form can evaporate any thin, liquid films on the waste form. Advective transport will cease if the heat from the waste form can evaporate the small seepage flux onto and into the waste package. The potential for evaporation to eliminate radionuclide transport is conservatively ignored in the current EBS abstractions. This is a minor conservatism for diffusive releases with the nominal (unintruded) scenario. The earliest breach of any waste package occurs beyond 10,000 years and the mean waste package lifetime is tens of thousands of years. Thermal effects are most significant for the first few thousand years after repository closure. In this situation, diffusive transport begins after the main thermal pulse has dissipated throughout the local host rock, so evaporative effects on diffusive transport are probably modest at best.

  Thermal effects may have tremendous benefit for defense-in-depth and degraded barrier sensitivity studies, particularly those in which both the waste package and drip shield are degraded.

- **Diffusion coefficient is based on a bounding abstraction**–The free water diffusion coefficient for all radionuclides is based on the self-diffusivity of water at 25°C, D_0. This approach provides a bounding value for the free water diffusivity of all radionuclide species relevant to the TSPA. Preliminary data indicate that the self-diffusivity of water, D_0, is conservative (larger) than the radionuclide-specific values by a factor of approximately 1 for KI (Grey 1972 [138541], Table 2, p-2) to 3.3 for Np(V)-carbonate (CRWMS M&O 2000 [129284], Table 7). This assumption enhances diffusive transport by a factor of 1 to 3.3, depending on the radionuclide and the chemistry of the aqueous solution within the waste package.

  The correction to the free water diffusion coefficient for the porosity of the (crushed tuff) invert is conservative by about 40 percent (CRWMS M&O 2000 [129284], Equations 6.4.1-9 to 6.4.1-10). In addition, the correction to the free water diffusion coefficient for saturation is being reevaluated for the TSPA-LA. This reevaluation is based on recent experimental data for the diffusion coefficient in crushed tuff at saturations representative of the invert. The magnitude of this reevaluation may reduce the diffusion coefficient by an order of magnitude or more, depending on the uncertainty in the experimental apparatus.

One aspect of the conceptual model for flow through the EBS is potentially nonconservative. All seepage that is diverted by the drip shield, denoted as flow pathway 3 in Figure 3.6-7, passes directly into the UZ. This is reasonable from the viewpoint of contaminant transport because the liquid diverted by the drip shield never comes in contact with the waste form and does not transport radionuclides. On the other hand, this diverted liquid may increase the general saturation level and advective velocity throughout the invert because of capillary forces.

The potential effect from capillarity in the invert is mitigated in the TSPA-SR model in two ways. First, the lifetime of the drip shields is generally less than the lifetime of the waste packages. The drip shields start to fail at 20,000 years and more than 50 percent have failed (one patch or more) by 40,000 years (see mean response curve in Figure 4.1-8). The waste packages, on the other hand, fail much more slowly, with less than 10 percent failed by 40,000 years (see Figure 4.1-9). The seepage through the waste package is therefore a much more limiting factor for radionuclide mobilization and transport than diversion around the drip shield. In other words, many drip shields have failed before a significant number of waste packages fail, so that there will be a substantial advective flux through the drip shields when the waste packages finally release radionuclides. The lifetimes of the drip shield and waste package therefore mitigate the potential nonconservatism in the EBS flow model.

The second mitigation arises because the liquid saturation in the invert is computed independently of the EBS flow model. The liquid saturation in the invert is an important parameter for diffusive transport through the invert. The invert saturation is determined by the abstraction of the multi-dimensional thermal-hydrologic calculations of the near-field environment. This abstraction incorporates the shedding of water by the drip shield and capillary effects in determining the time-dependent saturation in the invert. The abstraction is independent of the EBS flow model. The EBS flow model therefore has no direct effect on the parameters determining diffusive transport through the invert.

The flow assumption for liquid diverted by the drip shield is therefore anticipated to have at most a very minor impact on the TSPA-SR results. This conclusion will have to be reviewed for the TSPA-LA models if the Yucca Mountain site is designated by the President.

3.7 UNSATURATED ZONE TRANSPORT

UZ transport refers to the movement of radionuclides from the EBS of the potential repository, through the UZ, to the water table. Further movement of the radionuclides below the water table is discussed in Section 3.8. UZ transport is important as the first natural barrier to radionuclides that escape from the potential repository. UZ transport acts as a barrier by delaying radionuclide movement. If the delay is long enough that a given radionuclide decays significantly (i.e., if the transport time is large compared to the radionuclide half-life), then the UZ can have a large effect on decreasing the dose from that radionuclide at the biosphere.

The following subsections discuss the important processes and assumptions, the implementation for use in the TSPA simulations, how uncertainty and variability are treated in TSPA simulations, and some results and interpretations. The relationships of these submodels with each other and with other TSPA components are diagrammed in Figure 3.7-1. More detailed information on UZ transport, including full description and justification of the models and data, can be found in the *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774]) and associated AMRs.

3.7.1 Features, Processes, and Conceptual Model

The UZ water flow provides the background on which UZ transport takes place. The radionuclides can be carried along with the water, and can move within the water, by several mechanisms. As such, all of the UZ flow features and processes discussed in Section 3.2 affect UZ transport as well. This section discusses additional features and processes that affect transport of radionuclides. Some of the basic concepts for UZ transport are pictured in Figure 3.7-2. As shown in that figure, transport through welded tuff and nonwelded tuff tend to be rather different, with transport through fractures dominating in welded tuff and transport through pores in the rock matrix dominating in nonwelded tuff. Additionally, the existence of zeolitic alteration in some regions has an important effect on radionuclide transport, both because the zeolitic tuff has low permeability that affects the water flow, and because the zeolites have chemical properties that cause them to interact with many radionuclides.

Features, events, and processes that are concerned with UZ flow and transport are discussed in detail in *Features, Events, and Processes in UZ Flow and Transport* (CRWMS M&O 2000 [142945]). Many of those FEPs are related to climate, infiltration, and UZ flow. Such FEPs are, of course, relevant to UZ transport as well, but they are discussed in Section 3.2. Eleven of the primary FEPs are concerned strictly with radionuclide transport, and several others are concerned with both flow and transport issues. Many of the FEPs are standard processes that are fully included in the TSPA model, including four that relate strictly to radionuclide transport, such as "matrix diffusion in geosphere" (Section 3.7.1.2) and "colloidal transport in geosphere" (Section 3.7.1.5). Other FEPs are not included in the TSPA model (i.e., they are excluded), in some cases because it is conservative to neglect them (e.g., "radionuclide solubility limits in the geosphere"; CRWMS M&O 2000 [142945], Section 6.8.11), and in other cases, because they are argued to have insignificant impact on potential repository performance (e.g., "gas transport in geosphere"; CRWMS M&O 2000 [142945], Section 6.6.5). A complete list of primary FEPs and their status (whether included or excluded) is given in Appendix B.

Radionuclides can migrate in groundwater as dissolved molecular species or by being associated with colloids. Five basic processes affect the movement of dissolved or colloidal radionuclides: advection, diffusion, sorption, hydrodynamic dispersion, and radioactive decay (CRWMS M&O 2000 [145774], Section 3.11.2). Sorption is potentially important because it slows, or retards, the transport of radionuclides. Diffusion of radionuclides out of fractures into matrix pores is also a potential retardation mechanism because matrix transport is generally slower than fracture transport. However, sorption and matrix diffusion have less effect on colloids, so radionuclides can be more mobile on colloids than if dissolved in the water. One aspect of potential significance with respect to chain decay is that daughter products may have significantly different sorption behavior than the parent radionuclide, thus affecting transport. The radionuclides and radioactive decay chains modeled in the TSPA are described in Section 3.5.1.

3.7.1.1 Advection

Advection is the movement of dissolved or colloidal material along with the bulk flow of a fluid, which in this case is water. In many of the hydrogeologic units, advection through fractures is expected to dominate transport behavior. Advection through fractures is fast because of high permeability and low porosity, with few opportunities for radionuclides to interact with rock matrix. A few of the hydrogeologic units have much larger matrix permeability and are expected to capture most of the fracture flow by advection from the fractures to the matrix, causing much slower transport velocities and closer contact of the radionuclides with the matrix. Advective transport pathways are expected to be predominantly downward except in areas of perched water, where there is significant lateral flow. Flow that is diverted laterally ultimately finds a pathway to the water table through more permeable zones, which may be faults.

As discussed in Section 3.2.3, a dual-permeability model is used to represent mountain-scale UZ flow. The same concept is used to model radionuclide transport—a dual-continuum model, in which fractures and matrix are distinct interacting continua that coexist at every point in the modeling domain. Each continuum is assigned its own transport properties in addition to having its own hydrologic properties. The properties can vary spatially among hydrogeologic units.

Since mountain-scale UZ flow is represented as a sequence of steady states (Section 3.2.3.1), the flow field is abruptly changed from one to another at the time of a climate change. The transport calculation then continues with the new flow field. In addition to the change in the flow field, the location of the water table can also be changed abruptly at the time of climate change. The water table for the future climates (monsoon and glacial transition) is taken to be 120 m (390 ft) higher than the present-day water table (Section 3.2.3.1). When the water table rises with a climate change, the radionuclides in the UZ between the previous and new water-table elevations are immediately moved to the SZ.

3.7.1.2 Matrix Diffusion

Diffusion is the movement of dissolved or colloidal material because of random motion at the molecular scale. Diffusion results in mass flux at the macroscopic scale when the concentration is not uniform. Diffusion in the direction of transport is not an important mechanism for large-scale radionuclide transport, because other mechanisms (e.g., advection and hydrodynamic dispersion) lead to much faster radionuclide transport. However, diffusion can play an important role in radionuclide exchange between fractures and the rock matrix (Figure 3.7-3). Diffusion from fractures to the rock matrix can slow the advance of radionuclides undergoing advective transport through fractures (CRWMS M&O 2000 [141418], Section 6.1.3).

The effective diffusion coefficient is reduced by tortuosity effects (i.e., diffusion is slower through a tortuous network of matrix pores than when it is through pure water). The diffusion coefficient is also a function of the size of the diffusing molecules or particles. It has been found in laboratory measurements that anionic species such as pertechnetate (TcO_4^-, the predominant aqueous species of technetium) have lower diffusion coefficients than cationic species (CRWMS M&O 2000 [145774], Section 3.11.3.2). Colloids are much larger than these molecules and so have even smaller diffusion coefficients. Because of this, and possible size-exclusion effects, matrix diffusion is neglected for colloids.

The incorporation of matrix diffusion in the UZ transport model is simplified by neglecting flow in the matrix continuum within the diffusion calculation. This allows use of an analytical solution for matrix diffusion (CRWMS M&O 2000 [141418], Section 6.1.3). In this method, radionuclides are not transferred from the fracture continuum to the matrix continuum, but rather the radionuclides stay in the fracture continuum and transport at a slower rate that takes into account the fraction of time that diffusion would cause them to spend in the matrix pores. The fracture properties (spacing, fracture-matrix interface area) are modified to take into account that only some fractures are actively flowing and transporting radionuclides (CRWMS M&O 2000 [141418], Section 6.2.1). It has been shown that this approach has faster transport times relative to a dual-continuum implementation that includes the matrix flow (CRWMS M&O 2000 [134732], Section 6.2.5).

3.7.1.3 Sorption

Sorption is a general term for describing a combination of chemical interactions between the dissolved radionuclides and the solid phases (Figure 3.7-4); the solid phases involved can be either the immobile rock matrix or colloids. Possible interactions include surface adsorption, precipitation, and ion exchange. However, the sorption approach does not require identifying the specific underlying interactions. Instead, batch sorption experiments are used to identify the overall partitioning between the aqueous and solid phase, characterized by a sorption coefficient K_d. The strength of the sorption is a function of the chemical element, the rock type involved in the interaction, and the geochemical conditions of the water contacting the rock. The linear sorption model is used, which means that sorption is taken to be proportional to radionuclide concentration. Sorption reduces the rate of advance of a concentration front in advective and diffusive transport and amplifies the effects of matrix diffusion through its effect on the concentration gradient. The sorption coefficient can be combined with other terms in the transport equations to give an effective retardation factor.

The surfaces of fractures are often lined with minerals that differ from the bulk of the rock matrix and may be capable of sorbing some of the radionuclides. However, there has been limited characterization of the distributions of the fracture-lining minerals and sorptive interactions with these minerals. For these reasons, it is conservatively assumed that there is no sorption in the fracture continuum.

Although many different mineral types are present in Yucca Mountain, for purposes of characterizing sorption, the rocks are divided into three basic types: devitrified tuff, vitric tuff, and zeolitic tuff. The welded tuff is devitrified (this includes most of the Topopah Spring welded layers). The Topopah Spring basal vitrophyre is vitric tuff. The nonwelded tuff is either vitric or zeolitic depending on whether it underwent an alteration process during the original cooling period when the rock was formed. (This includes, in particular, the Calico Hills nonwelded layers.) There is more zeolitic tuff under the northern part of the potential repository and more vitric tuff under the southern part. It is important to note that the amount of sorption under actual transport conditions is not only a function of the sorptive strength, K_d, but also depends on the transport paths. The majority of the flow is through fractures in the devitrified tuff, and flow tends to bypass the zeolitic tuff because of its low permeability (Section 3.2.3). Thus, sorption is most effective in the vitric tuff. Its effectiveness in the devitrified tuff is tied to the effectiveness of matrix diffusion in pulling radionuclides out of fractures and into the rock matrix.

In TSPA simulations, the sorption characteristics of the rock are taken to be constant in time. Changes in sorption (or other transport properties) brought about by potential repository-induced thermal effects or climate change have been considered and found to be insignificant (CRWMS M&O 2000 [142945], Section 6.3.8, several subsections of Section 6.8).

3.7.1.4 Hydrodynamic Dispersion

Hydrodynamic dispersion refers to the spreading of radionuclides as they transport, caused by localized variations in the flow field and by diffusion (Figure 3.7-5). These effects spread the radionuclides both along and transverse to the average flow direction (referred to as longitudinal and transverse dispersion). Dispersion smears sharp concentration gradients and reduces the amount of time required for the initial arrival of low concentration levels of an advancing concentration front at a particular location (CRWMS M&O 2000 [145774], Section 3.11.2.3).

The dispersion resulting from variations in the flow field has an effect similar to diffusion, in which mass flux is modeled as being proportional to the concentration gradient. This proportionality, referred to as Fick's law, is the mathematical form used to represent hydrodynamic dispersion in the UZ transport model. The dispersion coefficient in the Fickian relationship is expressed as the product of a length scale called the dispersivity and the average linear water velocity. In principle, the diffusion coefficient would be added to this product, but the diffusion contribution to hydrodynamic dispersion is negligible. Dispersion is independently represented in both the fracture and matrix continua.

Dispersion is a way of including small-scale velocity variations in the transport model, but these small-scale variations are not very important to UZ transport, because they have less effect over long distances than the large-scale velocity variations that are explicitly modeled. Also, the explicitly modeled differences in transport velocity between fractures and matrix, and the transfer of radionuclides between them, introduce considerable dispersion into the transport simulations. Transverse dispersivities are normally small compared to longitudinal dispersivities. However, in the current implementation, dispersivity is simply taken to be isotropic. Any transverse spreading that does occur in the UZ is eliminated at the water table by starting the SZ transport at a small number of discrete points (Section 3.8.2.2).

3.7.1.5 Colloid-Facilitated Transport

Colloids, because they are small solids (from 1 nm to 10 μm), can interact with radionuclides through sorption mechanisms. Unlike sorption of radionuclides to the immobile rock matrix, however, radionuclides sorbed on colloids are potentially mobile. Therefore, colloids can facilitate radionuclide transport at a faster rate. Another form of colloidal radionuclide movement occurs when the radionuclide is an integral component of the colloid structure (Section 3.5.6). In this case, the radionuclide is irreversibly bound to the colloid, as compared to the reversible sorption mechanism. These two types of colloids will be referred to as reversible and irreversible colloids, according to whether the radionuclides are reversibly or irreversibly bound to the colloids (Figure 3.7-6).

It was mentioned in Section 3.7.1.2 that colloids diffuse much more slowly than dissolved radionuclides because of their much larger size. As a result, matrix diffusion is neglected for colloids. Colloids can, however, still move between fractures and the rock matrix advectively (i.e., simply moving with the water), as long as they are smaller than the matrix pores. Most of the pores are quite small in the welded and zeolitic tuffs, so many colloids remain in the fractures in those hydrogeologic units (CRWMS M&O 2000 [145774], Section 3. 11.3.4). Since transport in fractures is faster than transport in the rock matrix, this size-exclusion effect results in faster colloidal transport in those units. (Most flow and transport in those units is through fractures, though, so the result of this effect is not large.) In addition, a size-exclusion effect is possible at hydrogeologic unit interfaces. This exclusion is not applied to colloids transporting through fractures (fractures are relatively large compared to matrix pores), but it is applied to colloids transporting in the matrix from one hydrogeologic unit to another. In this situation, a portion of the colloids, corresponding to the fraction of them that are larger than the pores in the downstream unit, is stopped at the unit interface. This exclusion is taken to be a permanent filtration for irreversible colloids. It is not applied to reversible colloids, because the radionuclides can desorb from the colloids and continue to transport, even though the colloids may be stopped (CRWMS M&O 2000 [145774], Section 3.11.13.3).

Colloids may be temporarily detained at the fracture-matrix interfaces or sorbed to fracture walls (reversible filtration), and this interaction can be captured in the colloid transport model as a retardation factor for colloid transport in the fracture system, denoted R_c (CRWMS M&O 2000 [145774], Section 3.11.13.3; CRWMS M&O 2000 [141418], Section 5.3). The effective transport velocity is reduced by this factor. However, the only data available on this effect are for the SZ, so R_c is conservatively set to 1 in the UZ (which implies no retardation of the colloids).

For reversible colloids, radionuclides sorbed to colloids are assumed to be in equilibrium with radionuclides in solution. The ratio of the concentration of a radionuclide sorbed on colloids to the concentration in solution is represented by a parameter called K_c (CRWMS M&O 2000 [141418], Section 5.3). The K_c ratio is a function of the concentration of colloids and the sorption coefficient for the given radionuclide onto the given type of colloid. Because the radionuclides on reversible colloids are in equilibrium with radionuclides in solution, matrix diffusion can cause some effective slowing of the transport even though the colloids themselves do not diffuse into the matrix. (Diffusion of dissolved radionuclides into the matrix reduces the concentration in the fractures, which reduces the amount of radionuclides sorbed to the colloids and, thus, effectively slows the transport.)

Most radionuclides included in the TSPA model are strongly sorbing. Only carbon, technetium, iodine, uranium, and neptunium have little or no sorption (see Section 3.7.3). Without facilitation by colloids, transport of strongly sorbing radionuclides is extremely slow. Thus, colloid-facilitated transport is used for all strongly sorbing radionuclides in the TSPA model, including all radionuclides except for ^{14}C, ^{99}Tc, ^{129}I, ^{237}Np, and isotopes of uranium.

3.7.2 Implementation in the Total System Performance Assessment

Radionuclide transport in the UZ is implemented by using the residence-time transfer-function particle-tracking technique (CRWMS M&O 2000 [145774], Section 3.11.13.3; CRWMS

M&O 2000 [141418]). This technique is a cell-based approach in which particles move from cell to cell in a numerical grid. Particle locations within cells are not tracked, as they are in some particle-tracking techniques, but, rather, movement from cell to cell is computed probabilistically, based on transfer functions. The transfer functions are defined using analytical or semianalytical solutions of the transport equations, and represent probability distributions of the residence time (the amount of time that a particle resides in a cell). The probability that a particle will move to a given neighboring cell is proportional to the water flow rate to that cell. Only outflows are included in this calculation; particles are not moved to a cell if water flows from that cell to the current cell. A dual-continuum conceptual model is used for transport, so there is a network of fracture cells and a network of matrix cells, with each fracture cell connected to a corresponding matrix cell.

The connections between UZ transport and other TSPA model components are shown in Figure 3.7-7. The UZ transport model is directly coupled (i.e., dynamically linked) with the TSPA model. The UZ flow calculations are done ahead of time, and the flow fields are saved for use by the TSPA model (Section 3.2.3). During a TSPA simulation, radionuclide mobilization and transport through the EBS are calculated, and the radionuclide mass flux at the EBS boundary at each time-step is provided as the boundary condition for UZ transport (Section 3.6). The UZ transport model then provides radionuclide mass flux at the water table at each time-step as the boundary condition for SZ transport (Section 3.8).

The use of pregenerated flow fields implies the assumption of quasi-steady flow. That is, flow is modeled as a sequence of steady states (Section 3.2.3.1). The transport calculation itself is fully transient, with radionuclides moving downward from the potential repository as they are released. Each TSPA realization uses one set of flow fields; there are sets for low infiltration, medium infiltration, and high infiltration. Each set has three flow fields—for present-day, monsoon, and glacial-transition climates (Section 3.2). The water table is higher in the future climates. At the time of a climate change, when the water table rises (only at 600 years in the TSPA base case), any radionuclides in the interval below the new water table are immediately sent to the SZ for transport.

Releases from the EBS are computed for 30 environmental groups, which are based on infiltration, waste type, and seepage condition (Section 3.3.2). Since infiltration is important for UZ transport as well, radionuclides are released into the UZ at locations consistent with the environmental group from which they are released. Each environmental group is associated with one of five infiltration bins, based on the infiltration at each spatial location during the glacial-transition climate. The ranges for the bins are 0 to 3 mm/yr (0 to 0.1 in./yr), 3 to 10 mm/yr (0.1 to 0.4 in./yr), 10 to 20 mm/yr (0.4 to 0.8 in./yr), 20 to 60 mm/yr (0.8 to 2.4 in./yr), and 60+ mm/yr (2.4+ in./yr). Figure 3.7-8 shows the locations in the UZ transport model where those infiltrations occur for the three infiltration cases. This figure corresponds to Figure 3.3-3, which shows the infiltration-bin locations as they are modeled in the EBS. The two figures are somewhat different because of different discretization. The potential repository outline shown in this figure, and used in TSPA simulations for determining releases to the UZ, is not quite the same as the final SR design.

In order to avoid spreading out the radionuclides artificially, the injection of radionuclides into the UZ takes into account the number of waste packages that have failed within each of the five infiltration bins. If only one waste package has failed in a bin, the releases for that bin are put into a single UZ cell, sampled randomly from the cells in that bin (Figure 3.7-8). If two waste packages have failed, then releases are put into two randomly selected cells. This process continues for additional waste packages until the number of failed waste packages is equal to the number of cells in the bin. At that point, the releases are spread over all cells in the bin, and additional waste package failures cause no change to the release locations. For any number of failed waste packages in a particular bin, releases are always divided evenly among the cells that have been selected. Artificial spread of radionuclides in the UZ is further reduced by gathering the releases from the UZ into a few discrete locations at the water table for input to the SZ transport model. (For details on this procedure, see Section 3.8.2.2.)

Releases from the EBS are injected into the fracture continuum of the UZ transport model, so radionuclide transport through the UZ is initially through fractures. The radionuclides remain in fractures unless flow transfers from fractures to the rock matrix, in which case radionuclides will be advectively transported into the matrix continuum. (If a portion of the fracture flow transfers to matrix flow, then a portion of the radionuclides are carried into the matrix.) In addition, radionuclides can be transported into the matrix by diffusion, but in the UZ transport model this process is modeled as retarded fracture transport rather than transport into the matrix (Section 3.7.1.2). It should also be noted that flow tends to divert around open drifts, resulting in reduced flow in the region immediately below an emplacement drift (see, e.g., Figure 3.2-12). Such perturbations of flow caused by the presence of the drift are neglected in the model. Neglecting the flux "shadow" below the drift is conservative because the drier conditions at the EBS–UZ interface would increase transport times if included.

3.7.3 Treatment of Uncertainty and Variability

In the UZ transport model, spatial variability is included by use of a three-dimensional model that incorporates the appropriate geometry and geology. Temporal variability is included by using different UZ flow fields for different climate states. None of the other transport properties changes with time. Of course, the radionuclide source term also varies with time.

Uncertainty is included in the UZ transport model by defining uncertainty distributions for a number of input parameters. Values of these parameters for each TSPA realization are sampled from the distributions. Thus, each realization of the total system has a unique set of input parameters, each of which is within the range that is considered to be defensible. Normally, each realization is considered to be equally likely, unless importance sampling is used to emphasize some realizations (usually to increase the probability of sampling an unlikely event or parameter value).

Some of the uncertainty in UZ transport results from uncertainties passed to it by other models: uncertainty in infiltration and UZ flow from the UZ flow model; uncertainty in the number of failed waste packages from the waste package degradation model; and uncertainty in numerous EBS parameters and processes in the radionuclide source term received from the EBS transport model. The uncertainty distributions for parameters of the UZ transport model itself are summarized in Tables 3.7-1 and 3.7-2. Some key parameters that are treated as certain in the TSPA (i.e., that have single values rather than uncertainty distributions) are also listed.

Table 3.7-1. Sorption Coefficients for Unsaturated Zone Transport

Parameter Description	Distribution
K_d for Am and Th in devitrified tuff (mL/g)	Uniform; min = 100, max = 2,000
K_d for Am and Th in vitric tuff (mL/g)	Beta; mean = 400, COV = 0.2, min = 100, max = 1,000
K_d for Am and Th in zeolitic tuff (mL/g)	Uniform; min = 100, max =1,000
K_d for Pu in devitrified tuff (mL/g)	Uniform; min = 5, max = 70
K_d for Pu in vitric and zeolitic tuff (mL/g)	Beta; mean = 100, COV = 0.25, min = 30, max = 200
K_d for Np in devitrified tuff (mL/g)	Beta; mean = 0.3, COV = 0.3, min = 0, max = 1
K_d for Np in vitric tuff (mL/g)	Beta; mean =0.3, COV = 1, min = 0, max = 1
K_d for Np in zeolitic tuff (mL/g)	Beta; mean = 0.5, COV = 0.25, min = 0, max = 3
K_d for U in devitrified tuff (mL/g)	Beta; mean = 0.5, COV = 0.3, min = 0, max = 2
K_d for U in vitric tuff (mL/g)	Beta; mean = 0.5, COV = 0.3, min = 0, max = 1
K_d for U in zeolitic tuff (mL/g)	Beta; mean = 4, COV = 1, min = 0, max = 10
K_d for Pa in all units (mL/g)	Uniform; min = 0, max = 100
K_d for I, Tc, and C in all units (mL/g)	Not sampled; 0

Source: CRWMS M&O 2000 [145774], Table 3.11-1

NOTE: COV = coefficient of variation = standard deviation divided by mean; Am = americium; Th = thorium; Pu = plutonium; Np = neptunium; U = uranium; Pa = protactinium; I = iodine; Tc = technetium; C = carbon

Table 3.7-2. Additional Key Transport Parameters

Parameter Description	Distribution
Diffusion Coefficient for Am, Pu, Np, U, Pa, Th (m^2/s)	Beta; mean = 1.6×10^{-10}, SD = 0.5×10^{-10}, min = 0, max = 10^{-9} (CRWMS M&O 2000 [145774], Table 3.11-2; CRWMS M&O 2000 [148384], Table 6-78)
Diffusion Coefficient for I, Tc, C (m^2/s)	Beta; mean = 3.2×10^{-11}, SD = 10^{-11}, min = 0, max = 10^{-9} (CRWMS M&O 2000 [145774], Table 3.11-2; CRWMS M&O 2000 [148384], Table 6-78)
Dispersivity for both fractures and matrix (m)	Not sampled; 20 (CRWMS M&O 2000 [148384], Table 6-74)
Fracture aperture (mm)	Log-normal; geometric mean is different for each hydrogeologic unit, varying from 1.5 to 4.6 outside of fault zones and from 6.8 to 8.4 in fault zones (CRWMS M&O 2000 [141418], Table 3[a]); geometric SD = 1.9 (CRWMS M&O 2000 [141418], Section 6.2.1)
Fracture spacing (m)	Not sampled; different for each hydrogeologic unit, varying from 0.23 to 25 outside of fault zones and from 0.59 to 7.7 in fault zones (CRWMS M&O 2000 [141418], Table 3)
Colloid partitioning factor (K_c)	Log-normal; geometric mean = 3×10^{-3}, geometric SD = 10 (CRWMS M&O 2000 [147972], Table 15); only used for reversible colloids
Colloid retardation factor (R_c)	Not sampled; 1
Colloid size distribution (nm)	Not sampled; distribution of sizes from 1 to 450, median size approximately 75 (CRWMS M&O 2000 [148384], Table 6-88); only used for irreversible colloids
Fraction of colloids that can enter one matrix unit from another	Not sampled; function of colloid size and hydrogeologic unit (CRWMS M&O 2000 [141418], Table 6); only used for irreversible colloids
Fraction of colloids that can enter the matrix from fractures	Not sampled; different for each hydrogeologic unit, varying from 4% to 79% (CRWMS M&O 2000 [141418], Table 5); only used for irreversible colloids

NOTE: SD = standard deviation (geometric SD is 10 raised to the power equal to the standard deviation of the logs); Am = americium; Th = thorium; Pu = plutonium; Np = neptunium; U = uranium; Pa = protactinium; I = iodine; Tc = technetium; C = carbon

[a]The values listed as fracture apertures in this reference are actually half-apertures.

Sorption K_ds have been quantified using results of batch sorption experiments for different radionuclides and rock types (CRWMS M&O 2000 [141440], Section 6.4). Experiments with crushed-rock and whole-rock columns have also been performed for some radionuclides (CRWMS M&O 2000 [141440], Section 6.5). The sorption values in Table 3.7-1 can be used to separate the nine elements included in base-case TSPA simulations into three groups: technetium, iodine, and carbon have little or no sorption and are modeled using a K_d of zero; neptunium and uranium have small but nonzero K_ds, with expected values in all rock types less than or equal to 4 mL/g; and americium, plutonium, protactinium, and thorium are strongly sorbing, with expected K_d values of 50 mL/g or more in all rock types. Sorption coefficients for different rock types and different elements are not correlated in TSPA simulations.

The matrix diffusion values for radionuclides are based on measured diffusion coefficients of tritium and technetium (CRWMS M&O 2000 [145774], Section 3.11.3.2). For technetium, the predominant aqueous species is pertechnetate (TcO_4^-); the pertechnetate measurements are used to represent all anionic species. Measurements of tritium diffusion are used to represent all cationic species. Predictions of radionuclide transport for cationic radionuclides, using the diffusion coefficient for tritium and measured batch sorption coefficients, have been found to be conservative relative to measured diffusion behavior (i.e., slower than the actual diffusion) (CRWMS M&O 2000 [145774], Section 3.11.3.2). The uncertainty distributions for diffusion coefficients were defined to account for variations in rock type and water content.

Dispersivity is a property of the flow geometry that is determined by the structure of the fracture paths (for dispersion in the fracture continuum) or pore structure (for the matrix continuum). There are no measured data at Yucca Mountain that can be directly applied to determining dispersivity in the UZ. However, a value of 20 m (66 ft) over the approximately 300 m (1000 ft) of UZ travel distance is consistent with the dispersivity versus scale correlation of Neuman (1990 [101464]). Sensitivity studies indicate that radionuclide transport in the UZ is insensitive to dispersivity over a range from 0 to 75 m (250 ft) (CRWMS M&O 1998 [100364], Section 7.6.1.2.6).

In the UZ transport model, radionuclide transport is not sensitive to fracture spacing, so single values are used for the fracture spacing of each hydrogeologic unit. For consistency, however, that fracture spacing is adjusted to obtain the spacing between active fractures (CRWMS M&O 2000 [141418], Section 6.2.1). The fracture-spacing values listed in Table 3.7-2 are the initial geometric spacings, and do not include the active-fracture adjustment, which is calculated as a function of fracture saturation within the transport model. Fracture aperture does have potentially significant impact on UZ transport, so uncertainty distributions were developed for fracture aperture for each hydrogeologic unit. Fracture aperture was calculated from the ratio of fracture porosity and fracture-matrix interface area (CRWMS M&O 2000 [141418], Section 6.2.1). The underlying quantities (fracture porosity and interface area) were originally derived for the mountain-scale UZ flow model (CRWMS M&O 2000 [145771], Section 6.1). Fracture apertures were also estimated for the UZ flow model, but based on fracture frequency and permeability rather than fracture porosity and fracture-matrix interface area (CRWMS M&O 2000 [145771], Section 6.1.2.2). The permeability-based values are considered to be more appropriate for estimating flow properties, since permeability is one of the most important flow parameters. However, the porosity-based values are considered to be more appropriate for purposes of the transport model, because fracture porosity and fracture-matrix interface area are key transport parameters. The permeability-based fracture apertures are typically about a factor of 10 smaller than the porosity-based fracture apertures. The uncertainty in fracture aperture was derived from the uncertainty in permeability and fracture frequency, because more data are available for those quantities (CRWMS M&O 2000 [141418], Section 6.2.1).

The K_c factor for reversible colloidal transport is a product of the mass concentration of colloids in the groundwater and the effective K_d for sorption of the given radionuclide onto colloids. Because of the scarcity of information on colloids, and to simplify the colloidal-transport model, conservative values were used for both of these factors. A bounding value was used for the colloid concentration, and a conservatively high distribution of values was used for K_d (CRWMS M&O 2000 [147972], Section 6.14). In each TSPA realization, the same K_c value is used in the UZ and the SZ (Section 3.8.2.1.2). The colloid retardation factor, R_c, is conservatively set to 1 for all colloids in the UZ (Section 3.7.1.5).

As discussed in Section 3.7.1.5, two types of physical filtration are modeled for irreversible colloids: trapping of colloids at the interface between hydrogeologic units and exclusion of colloids from fracture-matrix transfer. Both filtration models require information about the distribution of pore sizes in the rock matrix. The pore size distributions were estimated from measured moisture retention curves (CRWMS M&O 2000 [141418], Section 6.2.5).

For transfer of colloids from one hydrogeologic unit to another in the rock matrix, the distribution of colloid sizes is compared to the distribution of pore sizes in the downstream unit, and only the fraction of colloids that are small enough is allowed to enter. The larger colloids become permanently trapped at the interface. The irreversible colloids are basically small pieces of waste form (Section 3.5.6) that have been able to move through the EBS into the UZ, so the distribution of sizes was based on data from waste form degradation tests (CRWMS M&O 2000 [147505], Section 6.3.1). The data include information on colloids between 6 and 450 nm, and a mean diameter of 120 nm is reported for test times out to 140 days. For UZ transport calculations, the lower limit of colloid size was extended down to 1 nm. (The reported mean colloid size was higher at a later time, but that information was not used. Note that larger colloids are more likely to get trapped, so using the smaller mean size is conservative.)

For exclusion of colloids at fracture-matrix interfaces, the same pore size distributions were used as for the hydrogeologic-unit trapping, but the model was simplified by calculating the exclusion fraction using a colloid size of 100 nm rather than considering a distribution of sizes (CRWMS M&O 2000 [141418], Section 6.2.5).

3.7.4 Results and Interpretation

Figure 3.7-9 shows breakthrough curves for transport of technetium and neptunium from the potential repository to the water table. Technetium and neptunium are used for illustration because technetium dominates dose results at early times and neptunium dominates dose results at later times (see Section 4.1). Technetium is modeled as a nonsorbing tracer, while neptunium has a small amount of sorption. Results are shown for all three climate states, using the medium-infiltration case. The breakthrough curves were generated by modeling a pulse release of particles at time 0, with the release spread uniformly throughout the potential repository.

The quantity plotted is the normalized cumulative breakthrough, which is the cumulative number of particles that have reached the water table at a given time, divided by the total number of particles released. In this and the following figures, it should be kept in mind that conservative approximations have been used in the UZ transport model (e.g., see Sections 3.7.1.2 and 3.7.2), which bias the breakthrough results toward lower transport times. The extent of the bias cannot be quantified without additional information.

Note that the results in Figure 3.7-9 are only for illustration. They are different from the base-case TSPA simulations in several respects:

1. In TSPA simulations, the present-day climate is used for the first 600 years, then the monsoon climate from 600 to 2,000 years, followed by the glacial-transition climate for the balance of the simulation (Section 3.2.1.4).

2. In TSPA simulations, radionuclides are released from selected potential repository locations rather than being spread over the entire potential repository (Section 3.7.2). The breakthrough curves in the figure can be thought of as the average over all potential repository locations.

3. The breakthrough curves in the figure were generated assuming no radioactive decay. However, ^{99}Tc and ^{237}Np have long enough half-lives (200,000 years and 2 million years, respectively) that inclusion of radioactive decay would only affect the portion of the plots at very late times.

4. In TSPA simulations, the water table is raised by 120 m (390 ft) for monsoon and glacial-transition climates (Section 3.2.3.1). The simulations in Figure 3.7-9 were simplified by keeping the water table fixed at its present-day location. However, it has been shown that water table rise does not greatly affect the UZ transport time (CRWMS M&O 2000 [134732], Section 6.2.4).

5. The neptunium sorption coefficients used for this illustration are somewhat higher than those used for the TSPA simulations, and the matrix-diffusion assumptions were somewhat different (see CRWMS M&O 2000 [134732], Section 6.1.1), for details of the parameter values used for Figure 3.7-9). The higher sorption makes the neptunium transport times longer than in the actual TSPA simulations, but the figure is still useful as an illustration of the effect of a small amount of sorption. Technetium is modeled as nonsorbing in all UZ hydrogeologic units (Table 3.7-1).

The breakthrough curves in Figure 3.7-9 show that the transport-time distribution is bimodal, with approximately 40 percent of the particles arriving at the water table relatively quickly and the rest spread out over a long time. The fast part of the breakthrough curves represents particles that were able to travel very quickly (in fractures) from the potential repository to the water table, with perhaps a little delay because of matrix diffusion. The slow part of the breakthrough curves represents particles that spent at least part of their time transporting through the matrix. The breakthrough time for 50 percent of the particles varies from about 300 years to 2,000 years for nonsorbing technetium and is about a factor of 10 longer for neptunium. (As discussed above, neptunium sorption is lower in TSPA simulations, so the increase in transport time is probably less than a factor of 10) The figure shows that transport is significantly slower under the present-day climate, but there is little difference between the transport times for the monsoon and glacial-transition climates.

UZ transport for the three infiltration cases is illustrated in Figure 3.7-10, which shows breakthrough curves for all three cases for the glacial-transition climate. The five differences from the TSPA base case discussed above apply also to this figure; the medium curves in Figure 3.7-10 are the same as the glacial-transition curves in Figure 3.7-9. The figure shows that there is a considerable spread in transport times for the three cases, and the medium-infiltration case is generally closer to the high-infiltration case than to the low-infiltration case. The breakthrough time for 50 percent of the particles varies from about 100 years to 4,000 years for nonsorbing technetium and, once again, is about a factor of 10 longer for neptunium. The low-infiltration case is much less bimodal than the others because its infiltration is low enough that fracture flow is not as pervasive.

The distribution of particles at the water table is illustrated in Figure 3.7-11. As for the previous figures, this plot was generated by modeling a pulse release of particles spread uniformly over the potential repository area (the red outline). For this figure, transport of hypothetical nonsorbing, nondiffusing particles was modeled to focus on the water flow paths. The medium-infiltration, glacial-transition flow field was used, with the water table raised to an elevation of 850 m (2,800 ft), up from approximately 730 m (2,400 ft) for present conditions. In the northern part of the potential repository, particles do not transport vertically downward, but, rather, are diverted laterally at the perched water and then drain down the Drill Hole Wash fault (see Figure 3.2-10 for identification of some of the faults in the model). Concentration of particles along the Ghost Dance fault is also visible in the figure, but many of the particles do transport nearly vertically downward in the southern part of the potential repository. This figure can be compared to the right-hand plot in Figure 3.2-8, which shows the distribution of water flux at the water table. (Figure 3.2-8 is for present-day climate rather than glacial-transition climate, but the features are qualitatively the same.) As discussed in Section 3.2.3.4, the difference in behavior between the northern and southern parts of the model domain occurs because the Calico Hills nonwelded tuff is mostly zeolitic in the north, leading to a low-permeability zone and extensive perched water in the simulation (CRWMS M&O 2000 [145774], Figure 3.7-9).

A final example of UZ radionuclide transport times is given in Figure 3.7-12. The TSPA model was used to generate this figure, including all the parameter uncertainty distributions as shown in Tables 3.7-1 and 3.7-2. One hundred realizations were simulated, with each realization having a different set of transport parameters, sampled from the uncertainty distributions. In each realization, a pulse of radionuclides was released at time zero, spread uniformly over the potential repository area. The curves shown in the figure are the average breakthrough curves over all 100 realizations. Some realizations have faster than average transport, and others have slower than average transport. Results are shown for technetium and neptunium, as before. In addition, plutonium transport results are shown for the two types of colloid-facilitated transport: reversible and irreversible attachment to colloids. These simulations include the climate changes at 600 and 2000 years. The transport behavior clearly changes at 600 years because of the change from present-day to monsoon climate. A small change in slope is also observable at 2000 years (when the climate changes from monsoon to glacial-transition).

It can be seen in Figure 3.7-12 that the irreversible colloids have the fastest transport in the model, with 50 percent of the irreversible colloids reaching the water table in about 500 years. The effects of physical filtration are noticeable, as the breakthrough curve for irreversible colloids levels off at a normalized breakthrough of about 90 percent. This occurs because approximately 10 percent of the irreversible colloids have been trapped at unit interfaces (on average). In comparison, the reversible colloids have the slowest transport in the model, taking about 300,000 years to reach a 50 percent breakthrough. And only 70 percent of the reversible colloids reach the water table within 1 million years. Technetium reaches 50 percent breakthrough in a little over 1,000 years, and neptunium reaches 50 percent breakthrough in about 3,000 years, with some particles taking much longer for both.

3.8 SATURATED ZONE FLOW AND TRANSPORT

The SZ at Yucca Mountain is the region beneath the ground surface where rock pores and fractures are completely saturated with groundwater. Groundwater is an important natural resource in Nevada and a possible avenue by which radionuclides, leaking from a potential repository at Yucca Mountain, could affect inhabitants of the region. The upper boundary of the SZ is called the water table. The potential repository is planned to be located approximately 300 m above the water table in the UZ.

As on the surface, underground water flows from higher levels to lower levels. Based on the water level observed in wells, groundwater in the vicinity of Yucca Mountain flows in a generally north-to-south direction. The recharge (input of water to the groundwater system) is predominantly from the highlands to the north, and the predominant outflows are pumping wells in the Amargosa Valley to the south. Of particular importance to the SZ component of the TSPA is this region from the potential repository to the place where people live in Amargosa Valley (Section 3.9). Around Yucca Mountain, groundwater flows through fractured volcanic rocks (and, deeper, through carbonate rocks), while closer to Amargosa Valley, the groundwater flows through alluvial deposits. The water table is typically 100 to 300 m below the ground surface near YMP, although in the Amargosa Valley, the water table can be at the ground surface as spring discharge.

The major purpose of the flow and transport component of the TSPA for the SZ is to evaluate the migration of radionuclides from their introduction at the water table below the potential repository to the point of release to the biosphere (e.g., water supply well) (Figure 3.8-1). Radionuclides can move through the SZ either as solute (i.e., in the dissolved state) or associated with colloids (i.e., particles small enough to remain suspended indefinitely in water). The input to the SZ flow and transport calculations is the spatial and temporal distribution of mass flux of radionuclides from the UZ (Section 3.7). The SZ component outputs a mass flux of radionuclides and the concentration of radionuclides in the water used by a hypothetical farming community or the representative volume of groundwater is calculated by the TSPA model (Section 3.9). Radionuclide concentrations are used to evaluate compliance with the EPA's proposed groundwater protection standard and to evaluate the annual dose to the hypothetical receptor, who is an average member of the critical group of people assumed to be most at risk in the NRC standard (proposed 10 CFR 63.115 [64 FR 8640] [101680]), or is the reasonably maximally exposed individual in the EPA standard (proposed 40 CFR 197.21 [64 FR 46976] [105065]). The SZ component is included in several analyses directed at regulations concerning the safety of a potential repository at Yucca Mountain (Figure 3.8-2). For the individual protection standard, as addressed by the nominal-case and disruptive scenarios of TSPA, the SZ is a major pathway to the biosphere. The SZ pathway is also of interest in the Environmental Impact Statement, although at different lengths and time scales than for the individual-protection standard. For the groundwater-protection standard, radionuclide concentrations are determined for the SZ model domain. For the human-intrusion analysis, the SZ is the intact barrier between a damaged container and the biosphere. The SZ component requires different features to address these different analyses, including different radionuclides, distances, and time scales.

For TSPA-SR, two models of SZ flow and transport are used: a three-dimensional process level model that is used to calculate, in detail, flow fields and the transport of individual radionuclides important to dose, and a one-dimensional flow tube model that is used to calculate the transport of daughter radionuclides (radionuclides that form by the decay of other radionuclides) of lesser importance. Results from these models that are salient to the performance of a potential repository at Yucca Mountain involve the transport time from the vicinity of Yucca Mountain to the geosphere-biosphere interface, located 20 km away. Transport time through the SZ for dissolved, nonsorbing, nonreactive radionuclides such as ^{14}C can be very short, less than 100 yr.; however, the median transport time for the present-day climate is on the order of 600 yr. These short transport times are mainly caused in the volcanic rocks by fast transport through widely spaced flowing intervals, with limited interactions with water in the matrix, and in the alluvium by the lack of sorption to retard the migration. Transport time for dissolved, sorbing radionuclides such as ^{237}Np is typically much longer, on the order of thousands to tens-of-thousands of years. Radionuclides that transport associated with colloids such as ^{239}Pu also show similarly long or longer transport times, but a significant fraction of the calculations can show very short transport times. This wide range in radionuclide transport times is principally due to the uncertainty in parameters that define groundwater velocity, matrix diffusion, sorption, and colloid properties.

3.8.1 Saturated Zone Flow

The relationship between SZ flow and other components of TSPA, and the information that is contributed to TSPA, are diagrammed in Figure 3.8-3. The SZ flow subcomponent takes inputs from the UZ flow subcomponent and produces outputs, in the form of flow fields, for the SZ transport subcomponent. The SZ flow subcomponent incorporates a significant amount of geologic and hydrologic data taken from drill holes in the vicinity of Yucca Mountain.

The SZ flow system at Yucca Mountain is part of the Alkali Flat-Furnace Creek groundwater subbasin of the larger Death Valley groundwater flow system. Groundwater flows regionally from recharge areas at higher elevations in mountain ranges to the north and east, toward natural discharge areas at springs, and through evapotranspiration at playas (Figure 3.8-4). Significant quantities of groundwater are also currently being discharged from the regional SZ system by pumping in areas such as the Amargosa Valley.

Based on measured water levels in wells, the groundwater flow is generally to the southeast near the potential repository, with a transition toward the south and southwest farther south of the potential repository. Groundwater that has flowed beneath Yucca Mountain is probably captured at pumping wells 20 km or more to the south in the Amargosa Valley under present conditions.

Under predevelopment conditions (before pumping began in the Amargosa Valley) and for the current climatic state, natural discharge of groundwater from beneath Yucca Mountain probably occurred further south at Franklin Lake Playa (Czarnecki 1990 [100376], p. 1-12), although spring discharge in Death Valley is a possibility (D'Agnese, Faunt et al. 1997 [100131], pp. 64 and 69). Under past, wetter, climatic conditions, groundwater discharge probably occurred at locations nearer to Yucca Mountain within the Amargosa Valley.

Groundwater flow in the SZ below and directly downgradient of the potential repository at Yucca Mountain occurs in fractured, porous volcanic rocks at relatively shallow depths beneath the water table and in fractured carbonate rocks of Paleozoic age at much greater depths. At distances greater than about 10 to 20 km downgradient from the potential repository where the volcanic rocks thin and are overlain by alluvium, groundwater flows either through the alluvium or the deep Paleozoic carbonates.

Differences in hydraulic head measure the driving potential for groundwater flow that is inferred from water level measurements in wells, referred to as the hydraulic gradient. Near the Yucca Mountain site, hydraulic head in the deeper volcanic units and in the Paleozoic carbonate aquifer, based on one well, are generally higher than in the shallower SZ, indicating the potential for groundwater to flow upward. Variations in temperature and heat flow, measured in boreholes in the SZ, suggest significant redistribution of heat by vertical groundwater movement in some areas. These observations suggest that there is (an imperfect) confining unit separating the shallow volcanic flow system from the deeper flow system. Water levels in wells near Yucca Mountain indicate that north of the site is a region of possibly large hydraulic gradient (potentially 150 m/km) although an alternative interpretation is that the higher apparent heads in wells north of the site are the result of perched water. West of the site is a region of moderate hydraulic gradient, corresponding to a 45-m increase in water table elevation. Water level data indicate a small horizontal hydraulic gradient (0.1 to 0.3 m/km) immediately southeast of the site. Groundwater flow from the potential repository site for a distance of 5 to 8 km is apparently to the southeast toward Fortymile Wash. From there, the apparent direction of groundwater flow for about 20 km is to the south-southwest.

Variations in water table elevations have been directly monitored for a few decades and past variations have been inferred from geological and geochemical data. Recent water-level fluctuations in most wells have been small—on the order of a few tenths of a meter—primarily in response to barometric variations and Earth tides. Highly transient and longer-term variations in hydraulic head of a few meters to a few decimeters have been observed following earthquakes. Variations in water level have been greater in the Amargosa Valley area because of pumping. Significantly higher water-table elevations (80 to 120 m higher than current elevations) at Yucca Mountain have been inferred from the locations of nearby paleospring deposits and from geochemical and mineralogical evidence from the UZ at the site (CRWMS M&O 2000 [145738]). Higher water-table elevations in the geologic past were apparently associated with wetter climatic conditions.

The aquifer in volcanic rocks has been hydraulically tested at many of the wells near the Yucca Mountain site, although there are limited borehole data between approximately 10 and 20 km downgradient of the potential repository. Most of the available hydraulic data are from single borehole tests using constant discharge, fluid injection, pressure injection, and radioactive tracer methods. From these tests, estimates of hydraulic conductivity (a factor that determines the amount of water that can flow through a material) in the fractured volcanic rocks of the SZ generally range over three orders of magnitude, depending on the depth and the particular hydrogeologic unit. Wide ranges in permeability are typical of natural systems (Freeze and Cherry 1979 [101173], p. 27).

The apparent hydraulic conductivity values determined from multiple-borehole hydraulic tests at the C-well complex tend to be much higher, by about two orders of magnitude, than the values from single-borehole tests for the same intervals (Geldon 1996 [100396], p. 69). The C-hole complex is located approximately 2.5 km to the southeast of the potential repository. Multiple-borehole hydraulic tests generally yield results that are more representative of large-scale hydraulic conductivity of the aquifer, suggesting that the single-borehole tests elsewhere at the site may have significantly underestimated the effective hydraulic conductivity (and, thus, the groundwater flow velocity). Results from the multiple-borehole tests are used in the SZ flow modeling for TSPA-SR because the multiple-borehole tests are more conservative in terms of transport time. The use of the single-borehole hydraulic test results in TSPA-SR would result in longer transport times.

Measurements of flow in the deeper wells in the SZ near Yucca Mountain indicate that groundwater production in most of the wells occurred in a few discrete intervals within the volcanic units. For performance assessment calculations, these results suggest that most groundwater flow in the fractured volcanic units is through only a small fraction of the saturated thickness.

3.8.1.1 Features, Processes, and Conceptual Model Related to Saturated Zone Flow

The features, events, and processes (FEPs) concerning SZ flow that are addressed for Site Recommendation (SR) are presented in Appendix B. Justification of whether a FEP should be included in the TSPA-SR modeling or not is given in *Features, Events, and Processes in SZ Flow and Transport* (CRWMS M&O 2000 [137359]). FEPs not considered involve hydrothermal activity in the SZ, large-scale dissolution of the flow media, and water-table decline. FEPs included in the TSPA-SR modeling involve the presence of wells, saturated groundwater flow, water-conducting features, and other FEPs as described in *Features, Events, and Processes in SZ Flow and Transport* (CRWMS M&O 2000 [137359]). The remainder of this section contains a discussion of how features and processes deemed for inclusion in TSPA-SR are conceptualized.

Saturated groundwater flow in vicinity of Yucca Mountain can be estimated by knowing the porosity of the flow media, the hydraulic conductivity, and the recharge of water into the flow media. The intergranular porosity is a measure of the void volume in the media through which water can flow, and it is determined by measurements of fracture and matrix porosity. Of more interest is the porosity through which flow actually occurs—fracture porosity in volcanic rocks and effective porosity in alluvium—which is determined by analysis of the degree that flow is in preferential paths through the media. The hydraulic conductivity is a measure of how easily flow occurs, and it is also dependent on the geologic structure.

The geologic structure of the flow media in the vicinity of Yucca Mountain consists of stratified, faulted, and fractured volcanic rocks (Figure 3.8-5). The faults and fractures are several orders of magnitude more permeable than the tuff matrix; thus, almost all flow occurs in these features. However, not all of these features can be expected to be transmissive (because of incomplete connectivity and other heterogeneity), and, thus, flow tends to be in some fractured zones, here called flowing intervals as identified through borehole flow meter surveys. Note that all fracture zones do not transmit water. Flowing interval spacing is important to matrix diffusion. South of Yucca Mountain, flow transitions from volcanic rocks to alluvium. The alluvial deposits generally consist of gravels, sands, and silts. Flow is through the porosity within these deposits; however, it is possible that preferred flow paths still occur (e.g., through regions of more permeable material).

Large-scale stresses on the geologic structure can cause differences in permeability in different directions (anisotropy) in the flow media. Because of the dominant north-south direction of the major faults in the region, it is suspected that such an anisotropy in permeability exists in the vicinity of Yucca Mountain. One estimate of the ratio of permeability in the approximately north-south direction to that in the east-west direction is 5 to 1 (Winterle and La Femina 1999 [129796]). The importance of anisotropy to repository performance is that a more southward flow path would increase travel distances in the tuff and reduce the amount of flow in the alluvium (Ferrill, Winterle et al. 1999 [118941], p. 7). A reduction in the flow path length in the alluvium would decrease the amount of total radionuclide retardation that could occur for those radionuclides with greater sorption coefficients in alluvium than in fractured volcanic rock matrix. In addition, potentially limited matrix diffusion in the fractured volcanic units could lead to shorter transport times in the volcanic units relative to the alluvium. The SZ site-scale model incorporated two alternative cases, isotropic (no anisotropy) and anisotropic.

Groundwater enters the region around Yucca Mountain, either laterally, from recharge primarily to the north around Pahute Mesa and from the east through Rock Valley, or vertically, as recharge through the UZ (Figure 3.8-6). Of the groundwater directly under Yucca Mountain, the major components appear to be local recharge and underflow from the west and north (CRWMS M&O 2000 [141399], pp. 107 to 108). The most recent recharge data from Fortymile Wash (Savard 1998, [102213]) was incorporated into the TSPA-SR calculations. The significance of the recharge from Fortymile Wash to the flow system is uncertain; however, ongoing testing in the vicinity of Fortymile Wash is developing information to aid in determining the affects of recharge at Fortymile Wash.

So far, this discussion has concentrated on present conditions at Yucca Mountain. Future conditions of the groundwater flow system at Yucca Mountain are unknown but can be estimated from past changes in climate (Section 3.2) and observations of paleospring deposits and other geochemical and mineralogical evidence that implies a higher water table in the past. If the water table were 120 m higher under Yucca Mountain, and if discharge of the flow system were at the Stateline paleospring deposits (on the border of Nevada and California in the Amargosa Valley), the hydraulic gradient would be approximately 3 m/km greater than the present-day approximately 1 m/km. Further, if the hydraulic conductivity were the same in the elevated flow system, the groundwater flux would increase by a factor of four (groundwater flux is directly proportional to changes in hydraulic gradient). This estimate corroborates the regional-scale model, which derived an estimated increase in groundwater flux of 3.9 for past, wetter conditions in the region (D'Agnese, O'Brien et al. 1997 [100132]). A flux multiplier of 3.9 is used in the TSPA-SR for the glacial-transition climate. With this value, the SZ flow velocities are likely to be overestimated because the regional flow model calculated the 3.9 flux multiplier for a full glacial climate, and in TSPA-SR it is being applied to the drier glacial-transition climate. In this sense, the glacial-transition flux is considered to be reasonably conservative.

The ratio of the groundwater flux multiplier for the glacial-transition climate to the present-day climate is the same as the ratio of the glacial-transition unsaturated zone infiltration to the present-day UZ infiltration (when averaged over the entire UZ model domain). Thus, the groundwater flux multiplier for the monsoon climate is estimated as the same fraction as the monsoon UZ infiltration is to the present-day UZ infiltration—a value of 2.7 (Table 3.8-1).

Table 3.8-1. Climatic Alterations to Saturated Zone Flux

Climate State	Saturated Zone Flux Multiplier
Present-Day (0 to 600 yr.)	1.0
Monsoon (600 to 2,000 yr.)	2.7
Glacial Transition (2,000 to 10,000 yr.)	3.9

3.8.1.2 Three-Dimensional Saturated Zone Flow Model

The primary tool used in TSPA-SR to describe SZ flow is a numerical model formulated in three-dimensions. The three-dimensional SZ flow model has been developed specifically to determine the groundwater flow field at Yucca Mountain—the flow paths and groundwater velocities from the potential repository footprint, or outline, at the present-day water table to a distance 20 km downgradient, the approximate distance to the nearest domestic extraction of groundwater. The three-dimensional geometry allows explicit consideration of geologic structure and its effect on flow paths. The purpose of the three-dimensional SZ flow model is to calculate a library of flow fields, essentially maps of subterranean groundwater fluxes, with which SZ transport of radionuclides are calculated (Section 3.8.2). The domain and structure of the three-dimensional SZ flow model are illustrated in Figures 3.8-4 and 3.8-7.

The three-dimensional SZ flow model is implemented using the FEHM computer program (CRWMS M&O 2000 [145738]). This computer program uses a control-volume finite-element method to solve for groundwater flux. The groundwater flux solution for the SZ flow field in TSPA-SR is for steady-state flow through a single-continuum, porous medium. The flow system is confined; i.e., the elevation of the water table does not change. As the three-dimensional SZ flow model only considers steady-state flow, changes in flux associated with climate changes are handled in the TSPA implementation (Section 3.8.2.5).

The three-dimensional SZ flow model encompasses an area of 30 km east-west by 45 km north-south, to a depth of 2,750 m below the water table. The model grid is an orthogonal mesh with 500 by 500 m horizontal resolution and with variable vertical resolution (10 to 400 m). The hydrogeologic framework in the model is based on a refined version of the regional geologic framework model used by D'Agnese, Faunt et al. (1997 [100131], pp. 29 to 35). Nineteen different hydrogeologic units are represented in the model (transport actually occurs through seven of the units; the 19 units are needed to accurately determine the flow domain). Faults are represented in the model by offsets within the hydrogeologic units. Two linear, vertical features, with low permeabilities to the west and north of Yucca Mountain, are explicitly included to simulate the moderate and large hydraulic gradient regions, respectively.

Inputs are the groundwater flux at the lateral boundaries of the model domain, groundwater recharge flux at the water table boundary, and the heterogeneous permeability of the elements within the model domain (Figure 3.8-6). The flux at the lateral boundaries on the sides of the model domain approximately match values estimated from a regional-scale flow model that describes flow over a large portion of southern Nevada and parts of eastern California, from Pahute Mesa in the north and the Spring Mountains in the east, down to Death Valley (D'Agnese, Faunt et al. 1997 [100131]). The recharge flux boundary condition on the top of the model domain is taken from the SZ regional-scale flow model and the lower boundary of the UZ site-scale flow model (CRWMS M&O 1999 [130979]). Focused recharge along Fortymile Wash, consistent with measurements, is included as a specified flux. Groundwater flow is not allowed to occur across the bottom boundary of the model. Average thermal conditions are applied everywhere as a function of depth below ground surface, and permeability is assumed to be uniform within each of the 19 hydrogeologic units in the model domain. (A discussion of this assumption is found in *Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA* [CRWMS M&O 2000 [139440], Section 5.1]).

The three-dimensional SZ flow model has been calibrated by an iterative process, using automated parameter estimation methods and the simulated water-table elevations were compared with measured water-table elevations that fall in the model domain. There is agreement between the simulation results and most of the well measurements, particularly in the area downgradient of the potential repository. The differences between simulated and measured water table elevations are less than 5 m for shallow wells downgradient of the potential repository, within a 10 km distance. This 5-m residual corresponds to one percent of the total change in head of 500 m across the entire model domain. The direction of groundwater movement in this flow model is consistent with the conceptual model of the system and is evident in the plot of the flux vectors and potentiometric surface (water-table elevation) shown in Figure 3.8-8 (Figure 3.8-8 corresponds to the model domain, which is also indicated as a solid blue line in Figure 3.8-4).

It should be briefly mentioned that a one-dimensional model is also used in the SZ modeling for TSPA-SR, primarily for modeling the transport of radionuclide daughter products (Section 3.8.2.3). The one-dimensional model was implemented in the TSPA-SR calculation to account for decay and ingrowth because the radionuclide transport methodology used in the three-dimensional SZ site-scale flow and transport model is not capable of simulating ingrowth by radioactive decay.

The one-dimensional SZ flow model is implemented with the GoldSim software (Golder Associates 2000 [151202]) in the TSPA model as a series of "pipes." Average specific discharge along different segments of the flowpath is estimated using the three-dimensional site-scale model. The resulting values of average specific discharge are applied to the individual pipe segments in the one-dimensional transport model. The pipe segments are defined at distances of 5, 20, and 30 km from the potential repository. The one-dimensional flow model describes steady-state flow as does the three-dimensional SZ flow model. But while the convolution-integral method can allow piecewise transport solutions from the three-dimensional SZ transport model to be fit together to allow changes in flux to approximate climate change, the one-dimensional SZ transport model cannot adjust for changes in flux. Hence, only the glacial-transition climate is implemented in the one-dimensional SZ flow model. This approach would tend to reduce transport times of radionuclides and, in this respect, is conservative.

3.8.1.3 Treatment of Uncertainty and Variability

The three-dimensional SZ flow model used in TSPA-SR is the best available representation of the physical system, that is consistent with the existing data. However, variability is inherent in a natural system covering hundreds of square kilometers, and uncertainty is inherent where data are only taken at a finite number of points. For TSPA-SR, uncertainty and variability in the groundwater system is managed in the TSPA calculations. For the TSPA SZ component, the transport time of radionuclides is the important factor to potential repository performance. The key parameters affecting radionuclide transport time in the SZ are defined by probability distributions to capture uncertainty and variability. Most of the uncertainty and variability in transport time is caused by transport processes (e.g., matrix diffusion and sorption; Section 3.8.2.3). The primary flow process affecting transport time is groundwater advection (flux) and, in the TSPA calculations, all of the uncertainty and variability of the groundwater flow system is concentrated in the probability distributions defining two parameters: groundwater flux and hydrologic anisotropy (Table 3.8-2).

Table 3.8-2. Stochastic Parameters for Saturated Zone Flow

Parameter	Distribution Type	Distribution Parameters (Bounds)
Groundwater Flux	Uniform	[0,1] low—0 to 0.13—0.06 m/yr (near potential repository); medium—0.13 to 0.87—0.6 m/yr (near potential repository); high—0.87 to 1—6 m/yr (near potential repository)
Anisotropy Ratio (Volcanics)	Uniform	[0,1] isotropic—0 to 0.5; anisotropic—0.5 to 1

Source: CRWMS M&O 2000 [147972], Table 16

Groundwater flux is defined by three discrete cases—low, medium, and high. The magnitude (high is 10 times medium, and medium is 10 times low) and the weightings of these cases (the weightings determine the frequency each case is sampled during the TSPA calculations) is based on the SZ expert elicitation (CRWMS M&O 1998 [100353]) distribution for groundwater flux, which, in turn, is based primarily on uncertainty in hydraulic conductivity. Only the medium-flux flow field is calculated directly in the three-dimensional model. The low-flux flow field is calculated by multiplying all of the medium-case flux vectors by 0.1; the high-flux flow field is calculated by multiplying all of the medium-case flux vectors by 10.

Hydrologic anisotropy is defined by two discrete cases—isotropic and anisotropic. The isotropic case is calculated with the permeabilities of the hydrogeologic units set to be equal in all directions. The anisotropic case is calculated, with the permeabilities of the fractured geohydrologic units in the area, to the south and east of the potential repository, modified to reflect a 5:1 ratio of permeability in the north-south direction relative to the east-west direction.

In the TSPA calculations, either the isotropic case or the anisotropic case is selected at random for each realization.

Of some importance here are the features and process that are not considered to be uncertain or variable in TSPA SZ flow. Uncertainty in the extent of, and the location of boundaries between, the 19 hydrogeologic units is not included. In particular, the alluvium size is held constant for the flow calculations. (For the transport calculations, the size of the alluvium is adjusted for each realization, but it is assumed that the flux through the alluvium does not change.) The hydraulic conductivity, although it differs between each of the 19 geohydrologic units, is considered to be constant within each unit. Uncertainty in present-day flow paths is considered only with regard to hydrologic anisotropy. No uncertainty or variability in climate-change time or magnitude (i.e., the climate-change flux multiplier) is considered. And when a climate change does occur, no uncertainty or variability in flow paths or water table elevation (in the SZ) is considered. Inclusion of the full range of these uncertainties could result in a broader range of TSPA results.

3.8.1.4 Results and Interpretation of the Saturated Zone Flow Model

Six flow fields are simulated for TSPA-SR, one each for the low-, medium-, and high-flux estimates for both the isotropic and anisotropic permeability cases (CRWMS M&O 2000 [139440]). (A flow field consists of a set of groundwater flux vectors associated with the grid points of the three-dimensional model domain.)

Figure 3.8-8 shows the flow field calculated by the three-dimensional SZ flow model for the present-day climate, medium-flux, and isotropic case. The results show flow primarily from north to south through the model domain. In the vicinity of Yucca Mountain, flow is to the southeast. Toward Fortymile Wash, flow becomes more southerly. On the plot, the potentiometric-surface contours are spaced by 1 m around Yucca Mountain and 5 to 100 m farther away, hence the bunching of lines at the mouth of Crater Flat. The potentiometric surface lines reflect lines of equal water table elevation and these contours indicate the direction of flow. This calculated potentiometric surface is in good agreement with observed water table elevations in drill holes.

3.8.2 Saturated Zone Transport

The relationship between SZ transport and other components of TSPA, and the information that is contributed to TSPA, are diagrammed in Figure 3.8-9. The SZ transport subcomponent takes inputs in the form of radionuclide mass fluxes from the UZ transport component and flow fields from the SZ flow subcomponent and produces outputs, in the form of radionuclide mass fluxes, to the Biosphere component. The SZ transport subcomponent incorporates a substantial amount of laboratory and field data taken from a variety of sources.

Radionuclides released from a potential repository at Yucca Mountain into the groundwater would enter the SZ somewhere beneath the potential repository and would be transported first southeast, then south, toward the Amargosa Valley. The radionuclides could be transported by the groundwater in two forms: as solute (dissolved in the water) or associated with colloids (Figure 3.8-10). Solute typically consists of radionuclide ions complexed with various groundwater species, but still at a molecular size. Colloids are particles of solids, typically clays or silica fragments, or organics, such as humic acids or bacteria, that are larger than molecular size, but small enough to remain suspended in groundwater for indefinite periods of time. Colloids are usually considered to have a size range of between a nanometer and a micrometer. A radionuclide associated with a colloid can transport either attached to the surface or bound within the structure of the colloid.

Radionuclide transport in the SZ depends not only on the flow of groundwater but also on the type of media through which the water is flowing. In the volcanic rocks that compose the saturated media in the immediate vicinity of Yucca Mountain, groundwater flows primarily through fractures, next to a large volume of water being held in relative immobility in the rock matrix. Radionuclides would travel with the moving fracture water but, if dissolved, could diffuse between the matrix water and fracture water, depending on concentration gradients. This transfer between water in the fractures and water in the matrix is characteristic of a dual-porosity system and is modeled as such in TSPA-SR. Further from Yucca Mountain, flow transfers into alluvial media. Here, radionuclides would travel through the pores in the gravels, sands, and silts.

Also important to transport is the chemistry of the groundwater and the electrochemical affinity of the radionuclides and the media. The groundwater in the transport pathway is generally thought to be oxidizing (at least where there is substantial flow), and it is so considered for TSPA-SR. The electrochemical binding of substances (e.g., radionuclides) to the surface of other substances (e.g., the rock matrix and alluvium) is called sorption or adsorption. Sorption is usually considered to be a temporary or reversible process. Sorption usually serves to hinder the transport of radionuclides, except when the radionuclides are sorbed onto colloids.

Data that support this description of radionuclide transport in the SZ come from several sources. Laboratory experiments have been used to investigate diffusion and sorption of radionuclides using groundwater and volcanic rocks and alluvium from the vicinity of Yucca Mountain. Field tests have investigated the chemistry of the groundwater and the mineralogy of the SZ. A complex of drill holes, known as the C-wells, have been used to investigate flow and transport on the scale of tens of meters in the natural groundwater system. In particular, forced-gradient, cross-hole tracer tests, conducted at the C-well complex, have provided data on in situ transport of the nonradioactive surrogate solutes and synthetic colloids (CRWMS M&O 1997 [100328]; Geldon et al. 1997 [100397], p. Background-2). Test results indicate that tracers diffused from fractures into the rock matrix, and that sorption occurred. The results also suggest that flow may have occurred in both fractures and the rock matrix during the tracer tests. There was relatively good agreement between tracer test results and laboratory measurements of sorption coefficients (K_d) for transport of lithium. Lower recovery of microspheres, a uniformly sized surrogate colloid with a neutral surface charge, suggests significant filtration over the 30-m transport distance. For TSPA-SR, these results support the idea of accounting for matrix diffusion in fractured volcanic units.

Groundwater sampling from fractured volcanic tuff in the SZ near the Benham underground nuclear test site on Pahute Mesa has found low concentrations of plutonium associated with colloidal material. The interpretation is that colloid-facilitated transport of plutonium in the SZ may be relatively rapid (Thompson 1998 [100788], p. vii), on the order of at least 1,300 m in

28 years (Thompson 1998 [100788], p. 13). For TSPA-SR, this observation supports inclusion of colloid-facilitated transport for several radionuclides (see Section 3.8.2.1.2).

3.8.2.1 Features, Processes, and Conceptual Model Related to Saturated Zone Radionuclide Transport

A discussion of how FEPs are used in TSPA-SR, and how they are included or excluded based on low consequence or low probability, is given in Section 2.2.1 and Appendix B. The FEPs concerning SZ transport that are addressed for SR are presented in Table B-15 of Appendix B. Justification as to whether an FEP should be included in the TSPA-SR modeling or not is given in *Features, Events, and Processes in SZ Flow and Transport* (CRWMS M&O 2000 [137359]). For example, FEPs not considered in TSPA-SR involve radionuclide solubility limits in the geosphere, suspension of particles larger than colloids, isotopic dilution, and other FEPs as described in *Features, Events, and Processes in SZ Flow and Transport* (CRWMS M&O 2000 [137359]). FEPs included in the TSPA-SR modeling involve advection and dispersion, matrix diffusion, colloid transport, radioactive decay, and ingrowth. The remainder of this section contains a discussion of how the most important features and processes included in TSPA-SR are conceptualized.

For TSPA-SR, the most important aspect of transport in the SZ to potential repository performance is how quickly radionuclides move from the vicinity of the potential repository to the biosphere. Important processes that must be considered in describing radionuclide transport time in the SZ include advection, hydrodynamic dispersion, diffusion (in particular, matrix diffusion), and sorption (Figure 3.8-11). Advection is the transport of contaminants by flowing water. Hydrodynamic dispersion is the tendency for contaminants to travel at different velocities because of interaction with the flow media, and, thus, to spread out. Diffusion is movement of a molecule (or colloid) by Brownian motion, typically from a region of higher concentration to lower concentration. Matrix diffusion is diffusion from water in fractures into water in the rock matrix. Sorption is the capture of molecules on mineral surfaces by electrochemical forces. Colloids can also sorb to mineral surfaces, and this process is often called chemical filtration. It is possible that physical filtration could occur in the alluvium, but it has been conservatively assumed to not occur in the TSPA calculations. At this time, data (e.g., grain-size data) are insufficient to support filtration in the alluvium at the Yucca Mountain site. How specific radionuclides are transported through the SZ is dependent on whether they transport as solute or in association with colloids.

3.8.2.1.1 Solute Transport

In the volcanic rocks, advective transport of solute is conceptualized by the dual-porosity model. The rock matrix holds immobile water and the radionuclides are carried by water flowing in the fractures. Radionuclides traveling down a fracture can diffuse across the short distance of the fracture width and into the immobile water in the matrix pores, then diffuse back into the fractures. This matrix-diffusion process can slow the contaminant movement, but, more importantly, it allows radionuclides access to sorption sites in the rock matrix. Sorption can significantly retard the transport of a radionuclide in groundwater, to the point where some strongly sorbing radionuclides, such as americium, plutonium, and thorium, cannot transport significant distances as solute and are only modeled as transporting when associated with colloids.

Radionuclide transport time in the alluvium also depends on flux and porosity of the flow media. However, in the alluvium, advection is conceptualized as transport through a porous medium, and, in order to approximate preferred paths in the porous medium, the porosity is described using the effective porosity concept. The effective porosity is the fraction porosity of the total porosity through which the radionuclides are carried. Using an effective porosity conceptualization reduces transport time, as compared with using total porosity. In contrast to fractured media, transport in the alluvium is through the pores of the medium and the medium is available for sorption.

3.8.2.1.2 Colloid-Facilitated Transport

The greater the sorption potential of a radionuclide, the greater the possibility that the radionuclide transports in association with colloids. All strongly sorbing (^{243}Am, ^{242}Pu, ^{241}Am, ^{240}Pu, ^{239}Pu, ^{238}Pu, ^{232}Th, ^{230}Th, and ^{229}Th) and moderately strongly sorbing (^{231}Pa, ^{137}Cs, and ^{90}Sr) radionuclides considered in TSPA-SR are modeled as having their mobility assisted by colloids. Other radionuclides (^{228}Ra, ^{227}Ac, ^{226}Ra, and ^{210}Pb) are modeled as being in secular equilibrium with parents that are associated with colloids, and, therefore, are effectively modeled as being transported by colloids. Secular equilibrium is the state where the activity of the daughter radionuclide is equal to the activity of the parent. Certain radionuclides that are daughter products of parent radionuclides that are irreversibly sorbed onto colloids (^{237}Np, ^{238}U, ^{236}U, ^{235}U, ^{234}U, and ^{233}U) are modeled as solute (the solute component from the initial source and the parents reversibly sorbed onto colloids are modeled separately). These moderately sorbing, relatively soluble radionuclides are assumed to disassociate from colloids when the parent decays, perhaps because of alpha recoil. Alpha recoil is the reaction of the nucleus when an alpha particle is emitted during radioactive decay. (This assumption is of little consequence, because these radionuclides are relatively mobile as solute.) Thus, most of the actinides and two of the fission products considered in TSPA-SR are being modeled as associated with colloids.

Colloid-facilitated transport can occur by two basic mechanisms. First, there are radionuclides that are permanently or irreversibly (on the time scales of interest) associated with colloids. These can be contaminants that are embedded within the structure of the colloids, such as a plutonium atom bound within the structure of a clay particle that formed by the degradation of a high-level waste glass waste form. Second, there are radionuclide contaminants that are temporarily or reversibly associated with colloids. These can be contaminants that are sorbed onto the surface of a colloid, such as a plutonium atom sorbed to the surface of a silica particle, or a particle of degraded spent nuclear fuel. Colloids that interact in reversible reactions with radionuclides are also called pseudo-colloids in the literature.

Radionuclides that are irreversibly associated with colloids (called irreversible colloids) are conceptualized as solute with a very low rate of diffusion in order to keep them restricted in the fractures and not allow matrix diffusion. Chemical filtration of colloids (often called filtration) is essentially sorption of the colloid, in the Yucca Mountain SZ, onto fracture surfaces or alluvium. Physical filtration in the alluvium, as stated earlier in Section 3.8.2.1, is not included in the TSPA-SR calculations.

The concept behind the model for radionuclides that are reversibly associated with colloids (called reversible colloids) is that they are in chemical equilibrium between (1) the dissolved state, (2) the state of being sorbed onto colloids, and (3) the state of being sorbed onto the aquifer material. The partitioning between these three states is defined by the Kc parameter—the product of the sorption coefficient for the radionuclide onto the colloid and the concentration of colloids available for sorption—and the Kd for the radionuclide on the matrix material. Conceptually, for the volcanic units, the Kc model keeps the radionuclides associated with colloids in the advective flow in the fractures more than if the radionuclides were solute. Conceptually, for the alluvium, the Kc model reduces the amount of the radionuclides associated with colloids that sorb onto matrix minerals compared to the amount that would sorb onto matrix minerals if the radionuclides were solute. This approach is conservative in the sense that the radionuclides associated with colloids result in faster transport times than if the colloids were solute.

The values of K_c in the SZ and UZ of the TSPA-SR probabilistic calculations are correlated. Filtration of reversible colloids is not considered, because even if the colloids filter, the radionuclides are free to dissociate and continue migrating (although filtration should retard movement to some extent). To simplify what could easily be an intractable problem, only one K_c parameter is used in the transport modeling for TSPA-SR. This value is based on sorption of americium onto waste form colloids in a low ionic-strength groundwater—a combination of factors which tends to maximize the mobility of the radionuclides. Also, only two sets of matrix K_ds are used in the K_c transport modeling for TSPA-SR—one for the highly sorbing radionculides (Am, Pu, and Th) and one for moderately highly sorbing radionuclides (Sr, Cs, and Pa)—and these K_ds are set to the minimum values that apply to all of the radionuclides in these categories. This combination of factors, maximized mobility and the lack of filtration in the alluvium, results in a conservative approach in terms of transport times.

3.8.2.2 Three-Dimensional Saturated Zone Transport Model

Transport in the SZ is modeled using a particle-tracking method. In concept, particles are released at a source point beneath the potential repository into the flow field produced by the three-dimensional SZ flow model. Then, as time progresses, the position of the particle is tracked down through the flow domain until it crosses a boundary (fence) at a distance of 20 km from the potential repository. The particle represents some mass of a given radionuclide. It also carries the properties of the radionuclide—specifically the sorption coefficient and whether it is associated with colloids or not. In the volcanics, matrix diffusion is accounted for by using a library of curves that specify transport time as a function of flow velocity, matrix porosity, effective diffusion coefficient, and flowing-interval spacing. The alluvium is considered to be a single porous medium, although preferred transport paths are still considered by using an effective porosity, which is less than the actual porosity. A set of particles are tracked in this manner in order to determine the breakthrough curves at the 20-km fence. Figure 3.8-12 shows particle tracks calculated by the three-dimensional SZ transport model.

The three-dimensional transport model is not used directly by the TSPA model. It is used to generate a library of breakthrough curves—distributions of transport times—that are used along with a time-varying source from the UZ to calculate the releases at the geosphere/biosphere boundary using the convolution integral method (Section 3.8.2.5). The reason for performing this involved procedure is that three-dimensional flow and transport modeling take too long for multiple-realization, multiple-radionuclide, and multiple-source-region TSPA calculations. The advantage of using a three-dimensional transport model is the accuracy gained by taking the real geometry of the groundwater system into account. The disadvantage is that radionuclide decay chains are not handled explicitly, necessitating a one-dimensional model to examine daughter products of some of the important decay chains (Section 3.8.2.3). Radionuclides modeled with the three-dimensional SZ transport model are ^{14}C, ^{90}Sr, ^{99}Tc, ^{129}I, ^{137}Cs, ^{234}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, and ^{243}Am (Figure 3.8-13).

The three-dimensional SZ transport model is implemented as follows. The library of breakthrough curves contains 3,200 separate curves—one curve for each radionuclide class (there are eight), for each source region (there are four), and for each TSPA realization (there are 100). The eight radionuclide classes with distinct transport characteristics are: ^{14}C, ^{99}Tc, ^{129}I, ^{237}Np, ^{238}U, radionuclides irreversibly sorbed onto colloids (plutonium and americium isotopes), and two cases of radionuclides reversibly sorbed onto colloids—those that are strongly sorbing onto matrix materials (i.e., americium, plutonium, and thorium isotopes), and those that are only moderately strongly sorbing onto matrix materials (i.e., ^{90}Sr, ^{137}Cs, and ^{231}Pa). Eight radionuclide classes are chosen in order to span the range of the transport characteristics for the radionuclides of interest. The four source regions are shown in Figure 3.8-14. The choice of four regions is arbitrary because one region could suffice, but four allows the possible effect of sensitivity to source location to be better discerned. (In fact, there is little sensitivity to source location with the TSPA-SR repository footprint.) Within each of the four regions, a source point is picked at random for each realization, and all the radionuclides that enter the region from the UZ are concentrated into the source point. (Note that this point-source method negates the need to track lateral-dispersion effects accurately in the UZ). The 100 realizations include variability and uncertainty in the breakthrough curves. The choice of 100 realizations is based on balancing the difficulty in performing large numbers of calculations with the need for accuracy in covering the range of possible transport behaviors in the results. Parameters that are described by probability distributions, including flow parameters (Section 3.8.1.3) and transport parameters (Section 3.8.2.3), are sampled for 100 values, and 100 separate calculations are made with these sampled values.

To create a breakthrough curve for use in the TSPA model, 1,000 particles with the properties of a radionuclide class are released at the source point in a single pulse. These particles are tallied when they cross the 20-km fence and the simulation ends when all 1,000 particles are tallied. There is variation in the arrival times because of the stochastic treatment of transverse and longitudinal dispersion for each particle. For some realizations, not all particles cross the fence in a reasonable time. When this situation occurs, the results are conservatively normalized as if 1,000 particles crossed the boundary. The resulting breakthrough curve is, thus, the cumulative relative mass released. Changes in groundwater flux because of climate change are handled in the TSPA model by scaling the breakthrough curves. Radioactive decay is also handled in the TSPA model.

3.8.2.3 One-Dimensional Saturated Zone Transport Model

A one-dimensional SZ transport model is used in the TSPA modeling to account for decay and ingrowth during transport in some of the daughter radionuclides considered in SR. The one-dimensional model is incorporated directly in the TSPA model (GoldSim [Golder Associates 2000] [151202]) as a series of pipes as described in Section 3.8.1.2. The advantage of using the one-dimensional SZ transport model is that radionuclide masses can be directly accounted for; the disadvantage is that the flow and transport geometry is necessarily simplified. In general, the radionuclides considered with the one-dimensional SZ transport model are of lesser significance to individual dose, but of significance to groundwater protection and, in some cases, to the million-year EIS calculations (Figure 3.8-15). Radionuclides considered with the one-dimensional SZ transport model are ^{210}Pb, ^{226}Ra, ^{228}Ra, ^{227}Ac, ^{229}Th, ^{230}Th, ^{232}Th, ^{231}Pa, ^{233}U, and ^{235}U (Figure 3.8-13).

The one-dimensional SZ transport model is based on an analytic solution to the advective/dispersive equation, with terms for sources and storage. The transport media consists of two units—a generic volcanic unit (from beneath the potential repository to the alluvium, a distance that varies between 12 and 19 km, depending on the realization) and alluvium (for the 20 km regulatory distance, a length that varies between 1 and 8 km, depending on the realization). Only a single source region, in the middle of the repository footprint, is considered in the model. During a TSPA calculation, 100 realizations are calculated, with the same parameter samplings as the three-dimensional SZ transport model. Because the transport equation only considers steady-state flow, changes in groundwater flux, because of climate change, cannot be implemented, and, therefore, the flux for the Glacial Transition climate is used in the model for all simulation times (see Section 3.8.1.2).

3.8.2.4 Treatment of Uncertainty and Variability

Uncertainty and variability in radionuclide transport through the SZ is incorporated into the modeling primarily by defining key model input parameters with probability distributions. These parameter distributions are defined either realistically or, in the absence of sufficient data, conservatively. Key model parameters are those that are uncertain and could significantly affect results, by changing the mean or variance in the transport time. The uncertainty and variability contained in the probability distributions is reflected in the library of breakthrough curves that is generated for TSPA-SR (Section 3.8.2.6). The distributions are sampled to define parameter for the library of 100 SZ breakthrough curves input to the TSPA-SR calculations. The uncertainty and variability contained in the library of breakthrough curves is then reflected in the TSPA results by using sampled breakthrough curves when each of the TSPA realizations is calculated (Section 3.8.2.5). The remainder of this subsection deals with the uncertainty and variability within the key model input parameters.

The primary transport processes that influence radionuclide transport time are those that control groundwater velocity, interaction with the matrix (especially matrix diffusion), sorption, and colloid-facilitated transport. The stochastic parameters used to model these processes in the SZ transport model are presented in Table 3.8-3.

Table 3.8-3. Stochastic Parameters for Saturated Zone Flow and Transport

Parameter	Distribution Type	Distribution Statistics [Bounds]
Flowing Interval Spacing (Volcanics) (m)	Log normal	Mean $\log_{10}$=1.29; SD $\log_{10}$=0.43
Flowing Interval Porosity (Volcanics)	Log uniform	[-5.,-1.]
Effective Porosity (Alluvium)	Truncated Normal	Mean=0.18, SD=0.051 [0, 0.35]
Effective Diffusion Coefficient (m^2/s)	Log uniform	[-13.,-10.]
Longitudinal Dispersivity (All Units) (m)	Log normal	Mean $\log_{10}$=2.0, SD $\log_{10}$=0.75
K_d for Np (Alluvium) (mL/g)	Beta	Mean=18.2, SD=18.8, [0,100].
K_d for Np (Volcanics) (mL/g)	Beta	Mean=0.5, SD=0.5, [0,2].
K_d for I (Alluvium) (mL/g)	Uniform	[0.32, 0.63]
K_d for Tc (Alluvium) (mL/g)	Uniform	[0.27, 0.62]
K_d for U (Alluvium) (mL/g)	Uniform	[0., 8.]
K_d for U (Volcanics) (mL/g)	Uniform	[0., 4.]
K_d for Am, Pu, Th (for K_c model) (mL/g)	Uniform	[0., 100.]
K_d for Pa, Cs, Sr (for K_c model) (mL/g)	Uniform	[0., 50.]
Irreversible Colloids Retardation Factor (Volcanics)	Piecewise CDF	[1.06, 800.]
Irreversible Colloids Retardation Factor (Alluvium)	Piecewise CDF	$\log_{10}$ [0.0011, 6.32]
K_c for Am, Pu, Pa, Th, Cs, Sr (All Units)	Log normal	Geo Mean=3×10^{-3}; Geo SD=10.
Alluvium Northern Boundary	Uniform	[0., 1.] (See Text)
Alluvium Western Boundary	Uniform	[0., 1.] (See Text)

Source: CRWMS M&O 2000 [147972], Table 16

NOTE: SD = standard deviation

Groundwater velocity is the quotient of the groundwater flux and the porosity of the medium. The parameters that control groundwater velocity in the SZ transport model are groundwater flux, flowing interval porosity, and effective porosity. Groundwater flux is discussed in Section 3.8.1.3. Flowing interval porosity is the porosity of those fractures in the volcanics that actually support significant flow. The distribution is estimated from several sources, including the C-wells' tests. Effective porosity is the porosity of the alluvium that actually supports significant flow; it is estimated to be approximately half the estimated total alluvial porosity. Longitudinal dispersivity (a parameter quantifying the spreading of the concentration front that occurs because of hydrodynamic dispersion) can also shorten radionuclide transport times for the leading edge of the concentration front. The distribution is based on the SZ expert elicitation (CRWMS M&O 1998 [100353]).

Interaction of the radionuclides with the matrix in the three-dimensional SZ transport model is controlled in the volcanics by the effective diffusion coefficient, flowing interval spacing, and flowing interval porosity. The effective diffusion coefficient is based on laboratory experiments and estimates of the tortuosity of the pore paths in the volcanic matrix. The flowing interval spacing is the separation distance between fractured zones that transmit significant flow. Flowing interval spacing is based on flow meter surveys from drill holes in the Yucca Mountain vicinity. In the alluvium, the interaction with the matrix is assumed to be complete, within the bounds of the effective porosity.

Sorption K_d distributions are based on laboratory measurements, with an estimate of the variability in mineralogy and groundwater chemistry that the radionuclides might actually encounter. Recent measurements have shown that even radionuclides previously thought to be nonsorbing (^{99}Tc and ^{129}I) do adsorb to the alluvium. No sorption for the fracture surfaces is modeled, although there is sorption in the matrix for uranium and neptunium.

For colloids with irreversibly sorbed radionuclides, a filtration parameter provides a retardation to transport (CRWMS M&O 2000 [129286]). For the volcanics, the retardation distribution is based on the C-Wells' tests. The median retardation factor is approximately 160. For these colloids, the diffusion coefficient is also set to restrict them to the fractures. For the alluvium, the lower end of the distribution is based on filtration theory, and the upper end is based on field tests taken from the literature. The median retardation factor is approximately 25. (Because the retardation applies to the transport velocity, this resulting transport time is typically slower in the alluvium than it is in the volcanics).

For colloids with reversibly sorbed radionuclides, the K_c and K_d parameters are highly uncertain, and, as discussed above in Section 3.8.2.1.2, several assumptions tending to minimize transport times were made to bound the many possible combinations of radionuclides and colloid types. The sorption potential is based on estimates of americium sorbed to colloids from degraded waste forms. The colloid concentration is based on the ionic strength of groundwater from drill hole J-13 and is consistent with colloid concentrations measured in J-13 groundwater.

For the three-dimensional SZ transport model, the areal extent of the alluvium is parameterized by two values, one that restricts the western extent and one that restricts the northern extent, as outlined by the solid yellow line in (Figure 3.8-16). The transport path always goes through approximately 1 km of alluvium. The average is approximately 4 km, and the maximum is approximately 8 km of alluvium. For the one-dimensional SZ transport model, the length of the alluvium is randomly picked to be between 1 and 8 km.

Not shown in Table 3.8-3 is the set of parameters used to specify the source point locations in the four source regions (Figure 3.8-14). These parameters pick a point at random in each of the regions. They allow the variability in the source location, and the concomitant variability in the transport path, to be incorporated in the three-dimensional SZ transport model.

3.8.2.5 Integration of the Saturated Zone Component into Total System Performance Assessment-Site Recommendation

The three-dimensional SZ transport model is used to determine concentration breakthrough curves at a distance of 20 km (and other distances) for unit releases of radionuclides. Then, within the TSPA calculations, the convolution integral technique is used to combine the breakthrough curves with the time-varying radionuclide sources from the UZ. The result is the mass flux for a given radionuclide at the geosphere-biosphere interface. These radionuclide mass fluxes are used to calculate radionuclide concentrations in the water usage volume (in the biosphere component; Section 3.9). This radionuclide concentration is used to calculate dose.

The one-dimensional SZ transport model is incorporated directly in the TSPA model and does not require the convolution integral method. In this section, the convolution integral method is discussed. Also, incorporation of climate change is addressed.

The convolution integral method is a computationally efficient method that combines information about the unit response of the system, as calculated by the three-dimensional SZ transport model, with the radionuclide source history to calculate transient system behavior (Figure 3.8-17). The most important assumptions of the convolution method are linear system behavior (i.e., doubling mass input results in doubling of concentration) and steady-state flow conditions in the SZ. (A discussion of this assumption is found in *Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA* [CRWMS M&O 2000 [139440], Section 5.5, #13]). To use the convolution integral method, the first derivative of the cumulative breakthrough curve is used. Also, within the convolution integral method is an adjustment to the amount of a radionuclide transported to account for radioactive decay. And because ingrowth from radionuclide parents is not handled in the three-dimensional SZ transport model, nor the convolution integral method, ingrowth is indirectly accounted for by increasing the mass flux of a radionuclide entering the SZ according to its parent's decay rate and the time scale of the calculation.

The effects of climate change on radionuclide transport in the SZ were incorporated into the analysis by assuming instantaneous change from one steady-state flow condition to another steady-state condition in the SZ. Changes in climate state were assumed to affect the magnitude of groundwater flux through the SZ system but have a negligible impact on flowpaths. (A discussion of this assumption is found in *Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA* [CRWMS M&O 2000 [139440], Section 5.6, #16]). These effects were incorporated into the convolution method by scaling the velocity of radionuclide breakthrough curves proportionally to the change in SZ specific discharge. The scaling factors are presented in Table 3.8-1. As mentioned in Section 3.8.1.2, the groundwater flux could not be arbitrarily changed in the one-dimensional SZ transport model, and, thus, only the groundwater flux for the Glacial Transition climate is used in this model.

3.8.2.6 Results and Interpretation of the Saturated Zone Transport Model

Although the actual results of the modeling of the SZ are hidden in the TSPA-SR calculations by several layers of detail (two different models, the convolution integral method, and dispersion circumvented by calculation of radionuclide concentration in a water-usage volume), the contribution of the SZ to potential repository performance can be evaluated here by examining radionuclide transport times. The breakthrough curves for unit concentration for the eight radionuclide classes at 20 km from the potential repository are shown in Figure 3.8-18. These curves were generated using the median values of the stochastic parameters and the flow field for the present-day climate, with radionuclides released at time zero.

Differences in the median arrival times of different radionuclide classes are because of variations in sorption and whether or not the transport is facilitated by colloids. The slopes of the curves decrease over time (note the logarithmic time axis). This increase is related to longitudinal dispersion and matrix diffusion. The influence of matrix diffusion is especially apparent when comparing the curve for irreversible colloids, which do not undergo matrix diffusion, to uranium, which does undergo matrix diffusion. Matrix diffusion causes a long tail on the uranium breakthrough curve. The curves all end up at the same maximum flux because they are all based on 1,000 particles, and radioactive decay is not included, though it is included in the convolution integral method.

In Figure 3.8-18, the radionuclide with the shortest transport times is carbon, a nonsorbing solute. The arrival of the median particle is at approximately 600 years. (For the glacial transition climate, groundwater flux is increased by a factor of 3.9, and this median particle would arrive at approximately 150 years. Other parameters [e.g., those that influence matrix diffusion] can also cause this transport time to be much different—either much longer or shorter.) The next shortest transport times are for technetium and iodine, two elements with only a slight sorption potential only in the alluvium; even this slight sorption retards transport by a factor of two. More strongly sorbing elements, uranium and neptunium, have transport times that are retarded by an order of magnitude or more. Much of these long transport times are due to significant sorption in the alluvium. Radionuclides that transport associated with colloids show a wide range in transport times, although typically the transport times are relatively long (e.g., on the order of tens of thousands of years). Radionuclides that are irreversibly sorbed onto the colloids have the shortest transport times of the colloid-facilitated species, but the retardation caused by filtration is still significant, causing the median particle to arrive in approximately 10,000 yr. For the radionuclides that are reversibly sorbed onto the colloids, those with the lower sorption potential onto the matrix are the faster (median particle arrival in approximately 60,000 years) than those with the higher sorption potential, by approximately a factor of two. Note that these times would be delayed in the actual TSPA calculations by the waste-package lifetime and the transport time through the UZ.

To include parameter uncertainty in the TSPA, 100 breakthrough curves were simulated for each radionuclide class by sampling 100 values for each parameter from their respective probability distributions (Table 3.8-3). The impact of uncertainty in the three-dimensional SZ transport model for the SZ is illustrated in Figure 3.8-19. This figure shows the breakthrough curves for carbon and for irreversible colloids for all 100 realizations used in the base-case analyses. The breakthrough curves are for the groundwater flux for the Present Day climate. Transport times for the nonsorbing ^{14}C vary from less than 100 years to greater than 100,000 years. For the irreversible colloids, the transport times vary from less than 100 years to on the order of 1 million years. Again, radioactive decay is not included in the breakthrough curves. Note that over time, inclusion of radioactive decay would cause the breakthrough curves to reach lower maximum values. The variance in the transport times is directly related to uncertainty in key parameters: the groundwater velocity (a function of the flux and the flowing interval porosity and the effective porosity in the alluvium), the amount of matrix diffusion (a function of the diffusion coefficient and the flowing interval spacing), and the amount of retardation (a function of sorption, colloid filtration, and other colloid-related parameters). In general, uncertainty in transport of radionuclides associated with colloids is greater than uncertainty in transport of solute. Both, however, are influenced by uncertainty in groundwater flux.

Delay in radionuclide migration in the SZ is significant for transport times long relative to the 10,000-year proposed regulatory standard. For nonsorbing (or slightly sorbing) radionuclides, such as carbon and technetium, transport times in the SZ are typically less than 10,000 years. Transport times for plutonium and americium subject to transport by irreversible attachment to colloids and uranium are near 10,000 years for the median-value case. Neptunium and radionuclides subject to reversible attachment to colloids have transport times in the range of about 20,000 years to 100,000 years in the SZ expected-value case, indicating that they are of lesser importance in the context of a 10,000-year regulatory standard.

3.9 BIOSPHERE

The biosphere is that part of the Earth characterized by biological activity. It includes the soil, surface water, the air, and all living organisms. Living organisms, including humans, residing in the biosphere could be affected by radionuclide release from a potential repository at Yucca Mountain only if these contaminants reach the biosphere.

The biosphere component of TSPA-SR is designed to predict radiation exposure to a person living in the general vicinity of the potential repository if there is release of radioactive material after closure of the potential repository. The biosphere component includes the following features. The human receptor and the reference biosphere is as proposed by the NRC (proposed 10 CFR 63.115 [64 FR 8640 [101680]]), and also applies to the receptor and biosphere proposed by the EPA (proposed 40 CFR 197.15, 197.21, and 197.37 [64 FR 46976 [105065]]). Radionuclides transported in groundwater are mixed in the annual water usage of a hypothetical farming community as recommended in the supplementary information for proposed 10 CFR 63 (64 FR 8640 [101680], p. 8646). Radionuclide buildup in soils because of continuing periods of irrigation with contaminated water is considered in the analyses, and estimates of soil and radionuclide removal by erosion, leaching, crop removal, and radioactive decay are incorporated into overall dose calculations.

The biosphere is the last component in the chain of TSPA-SR modeling subsystem components. Upstream from the biosphere, there are two connections. One is for the groundwater irrigation scenario (nominal case), in which the biosphere is coupled to the SZ flow and transport model; and the other is for the disruptive scenario, in which the biosphere is coupled to the volcanic dispersal model (see Section 3.10 for a description of the biosphere modeling for the volcanic scenario). Figure 3.9-1 shows the major connections between the biosphere component and the other components in TSPA-SR. An overview of the biosphere component is presented in Figure 3.9-2.

GENII-S (Leigh et al. 1993 [100464]), a computer program accepted by regulatory agencies including the NRC and the EPA, for predicting radiation dose, is used to calculate radionuclide-specific BDCFs. The biosphere modeling does not explicitly perform the dose assessment for TSPA-SR. The biosphere modeling provides an estimate of the dose incurred by a receptor when a unit amount of a radionuclide reaches the geosphere-biosphere boundary. This estimate (the BDCF) is in the form of a probability distribution to reflect biosphere model uncertainty. In the TSPA model, when a concentration of a radionuclide in groundwater has been calculated (within the computer program a mass flux is calculated and converted to a concentration), the BDCF sampled from the distribution is used as a multiplier to convert the concentration into annual dose.

A comprehensive list of FEPs is used to distinguish the attributes of the biosphere model. A discussion of the biosphere FEPs is presented in Appendix B of this document. These FEPs have been screened to determine those that should be included in the biosphere modeling and those that should be excluded from consideration. FEPs excluded from consideration concern those related to long-term environmental changes (e.g., erosion and denudation, fluvial, aeolian, and lacustrine deposition, or human influences on the atmosphere), capillary rise above the water table, transport and mixing by surface water, burrowing animals, contamination of clothing, furniture, and pets, radon, and various inapplicable FEPs (e.g., marine processes, urban and industrial water use, and nonradiological toxicity). Twenty-two primary FEPs have been recognized for inclusion (in part or in total) in the biosphere modeling. These FEPs concern present-day human lifestyles, dietary habits, household activities, agricultural land uses, and bioaccumulation in plants and animals. These FEPs are included in the biosphere modeling and are further discussed in this section.

The primary result of the biosphere modeling for TSPA-SR is the construction of BDCF distributions, for both the groundwater-release scenario and the volcanic-ash-release scenario (Section 3.10). Additionally, sensitivity studies were conducted for the groundwater-release scenario to determine the most important parameters in the model that contribute the most to the variance in the BDCF distributions (i.e., the parameters that cause the most spread in the final BDCFs), as well as a pathway analysis to identify which exposure pathways are the most important contributors to the BDCFs. For almost all radionuclides, the majority of the dose could be attributed to two pathways in the groundwater-release scenario: drinking water and leafy vegetables. The parameters that contributed most to uncertainty in the BDCFs include: the crop interception fraction (the fraction of contamination in rainfall, irrigation water, or aerosols that adhere to plant surfaces) and the soil-plant transfer scale factor (a factor representing uncertainty in the amount of radionuclides taken up by plants).

3.9.1 Definition of the Receptor

For TSPA-SR, the receptor (the person who incurs a radiation dose) is the "average member of the critical group," where the "critical group resides within a farming community consisting of approximately 100 individuals, and exhibits behaviors or characteristics that will result in the highest expected annual doses" (proposed 10 CFR 63.115 [64 FR 8640 [101680]]). The receptor in TSPA-SR is also the reasonably maximally exposed individual as defined by the EPA (proposed 40 CFR 197.21 [64 FR 46976 [105065]]). (The primary quantitative difference between the two concepts is that the average member of the critical group consumes the average amount of drinking water for individuals in Amargosa Valley, 752.8 L/yr [Section 3.9.2.3], and the reasonably maximally exposed individual is prescribed to consume 2 L/day or 730.25 L/yr. For TSPA-SR, the receptor is characterized as drinking the average amount, as it is the greater of the two.) In the following subsections, the characteristics of the local environment, the critical group, and the receptor are defined.

Amargosa Valley Environment–Yucca Mountain is located within the sparsely populated region between the Great Basin and the Mojave Deserts in southern Nevada. The climate in the vicinity is arid to semiarid. The natural vegetation is predominantly desert scrub and grasses.

The nearest community in the direction of groundwater flow from the potential repository site is Amargosa Valley (Figure 3.9-3). Amargosa Valley is an area of approximately 500 mi^2, defined as a tax district by the Nye County commissioners in the early 1980s. The closest inhabitants to

Yucca Mountain are at a location known as Lathrop Wells within Amargosa Valley, approximately 20 km south of the potential repository at the intersection of US 95 and Nevada State Route 373. There are approximately eight inhabitants at this location. This area is the general site of the hypothetical farming community that has been proposed by the NRC (proposed 10 CFR Part 63 [64 FR 8640 [101680]]). The Amargosa Farms area, a triangle of land bounded by the Amargosa Farm Road to the north, Nevada State Route 373 to the east, and the California border running from the northwest to the southeast, is the closest agricultural area. It is approximately 30 km south of the potential repository.

The Amargosa Valley area is sparely populated and primarily rural in nature (Figure 3.9-4). The area supports a population of approximately 1270 in about 450 households (DOE 1997 [100332], Section 2.4). The area has a general store (Figure 3.9-4A), a community center, a senior center, an elementary school, a public library, a medical clinic, a restaurant, a hotel-casino, and a motel. Agricultural activity is directed primarily toward livestock feed production, but gardening and animal husbandry are common. Commercial agriculture in the Amargosa Valley farming triangle currently includes a diary operation (approximately 4,500 dairy cows) employing about 50 people, a catfish farm that sustains approximately 15,000 catfish, and a garlic farm that annually produces about one ton of garlic. However, alfalfa is the predominant crop produced in the Amargosa Valley (Figure 3.9-4B), and alfalfa and forage grasses comprise a major proportion of Nye County agricultural land (LaPlante and Poor 1997 [101079], p. 2-6). Approximately 1,800 acres are dedicated to alfalfa production, 30 acres to oats production, 80 acres are in pistachios, and 10 acres in grape vineyards. Water for all uses—domestic, agricultural, commercial, and mining—is taken from local wells, mostly privately owned.

Regulatory Considerations–Regulation proposed by the NRC (10 CFR 63.115 [64 FR 8640 [101680]]) establishes the characteristics of the critical group considered in the TSPA-SR analysis:

- The critical group shall reside within a farming community located approximately 20 km south from the underground facility (in the general location of U.S. 95 and Nevada State Route 373, near Lathrop Wells, Nevada).

- The behaviors and characteristics of the farming community shall be consistent with current conditions of the region surrounding the Yucca Mountain site. Changes over time in the behaviors and characteristics of the critical group including, but not necessarily limited to, land use, lifestyle, diet, human physiology, or metabolics, shall not be considered.

- The critical group resides within a hypothetical farming community consisting of approximately 100 individuals, and exhibits behaviors or characteristics that will result in the highest expected annual doses.

- The behaviors and characteristics of the average member of the critical group shall be based on the mean value of the critical group's variability range. The mean value shall not be unduly biased based on the extreme habits of a few individuals.

- The average member of the critical group shall be an adult. Metabolic and physiological considerations shall be consistent with present knowledge of adults.

Regional Survey of Inhabitants–In order to determine the characteristics of the population in the vicinity of Yucca Mountain, a survey was conducted of inhabitants residing within an 80-km grid centered on Yucca Mountain (DOE 1997 [100332]). Figure 3.9-5 shows the grid with the color coding indicating the population density and the number in each sector indicating the approximate number of permanent inhabitants within the sector. The survey was conducted in the spring of 1997 and was focused on the Amargosa Valley region, but included Beatty, Indian Springs, and Pahrump. The survey was designed to provide for more comprehensive survey representation of the inhabitants closer to Yucca Mountain. It was estimated that 13,000 adults reside in the survey area, with 900 adults (seven percent) residing in the Amargosa Valley. Over one thousand interviews were completed for the survey, including interviews of 43 percent of the households in the Amargosa Valley.

The survey followed the principles developed by the U.S. Office of Management and Budget (e.g., Subcommittee on Questionnaire Design 1983 [100483]). Underlying the entire project were total design method principles and interviewing standards promulgated by the Institute for Social Research, University of Michigan (Guenzel et al. 1983 [101072]) to maintain high response rates and accuracy. To ensure accuracy, the survey aimed at minimizing sample error and nonsampling error. The survey included Spanish language interviews to accommodate respondents whose primary language is Spanish. Measures were taken to compensate for subjects who were difficult to interview. Additionally, demographic information was used to compensate for any gender bias that may have arisen.

The survey was designed to permit an accurate representation of dietary patterns of inhabitants in the region. Of special interest was the proportion of locally grown foodstuffs that was consumed by local residents (i.e., irrigated with groundwater potentially contaminated in the future), and details of what food types were eaten on a regular basis. In general, a higher percentage of locally produced food is consumed by residents in the Amargosa Valley than by residents in the remainder of the survey area. Nearly 80 percent of the survey respondents reported consuming locally produced food of some type over the past year in the Amargosa Valley, while only about 57 percent did so in the remainder of the survey area. Thus, Amargosa Valley residents have food consumption habits that make them more susceptible to radionuclide intake through the ingestion pathway than do their immediate neighbors, supporting their designation as a likely population that includes the critical group. Nearly 88 percent of Amargosa Valley residents consumed well water, 79 percent did so in the remainder of the survey area. No person interviewed in Amargosa Valley fit the description of a subsistence farmer; i.e., no respondent consumed only locally grown foodstuffs.

The biosphere modeling required estimates of annual consumption of selected food groups in terms of weight. Although it was not feasible to collect this type of information directly through the survey, it was feasible to collect frequency information on food consumption. Therefore, data taken from tables compiled through national surveys on food intake (USDA 1993 [101089], pp. 18 to 29) were combined with information from the survey to produce estimates of annual quantities, in kilograms, of the various food groups consumed (DOE 1997 [100332], Section 3.6). Because the regulation proposed by the NRC (proposed 10 CFR 63.115 [64 FR 8640

[101680]]) prescribes that the receptor be based on the average member of the critical group, the mean values of the estimated annual consumption quantities for residents of Amargosa Valley are used in the biosphere modeling (see Section 3.8.2.3 and Table 3.9-41).

Lifestyle Characteristics–In addition to the regional survey, other sources of data have been utilized to describe the receptor. Related to lifestyle characteristics, 1990 Census Data (U.S. Census Bureau 1999 [135344]) have been examined to evaluate employment attributes. Based on existing employment classifications, the highest exposure due to outdoor employment activity would be for agricultural or construction workers. For the recreational attribute, outdoor activities on contaminated land have the highest potential for exposure. As with food consumption, the mean values of the lifestyle data are used in the biosphere modeling (see Section 3.8.2.3 and Table 3.9-1).

3.9.2 Biosphere Model

The biosphere model describes the movement of radionuclides through the environment to the receptor and the subsequent radiation dose that the receptor incurs. Here, the characteristics of the biosphere model are defined. A diagram of how the biosphere model fits within the TSPA effort is shown in Figure 3.9-6.

3.9.2.1 Features, Processes, and Conceptual Model Related to the Biosphere

The biosphere model is based on a conceptual description of the surface environment in the vicinity of Amargosa Valley, taking into account the FEPs (Appendix B of this document and above), the proposed 10 CFR Part 63 (64 FR 8640 [101680]), and internationally accepted practices in modeling the biosphere (e.g., BIOMOVS II 1996 [100363]). A complete description of the biosphere conceptual model can be found in the *Biosphere Process Model Report* (CRWMS M&O 2000 [151615], Section 3.1).

Conceptual Model–After the permanent closure of the potential repository, radionuclides could eventually leach to the underlying groundwater and, subsequently, migrate downgradient to a location beneath the critical group. The contaminated water could then eventually reach the biosphere through the pumping of well water.

For the nominal-case scenario, well withdrawal of groundwater is considered the source of water for drinking, irrigation, and other uses. The affected farming community is located 20 km south of the potential repository in the Amargosa Valley region. All radionuclides reaching the farming community in groundwater are assumed to be mixed in the volume of water that the community uses. The exposure pathways—routes taken by radionuclides through the biosphere, from the source to a receptor—are typical for a farming community in this environment. Farming activities usually involve more exposure pathways than other human activities in the Yucca Mountain region and can include ingestion of contaminated water and locally produced foodstuffs, as well as inhalation and direct exposure from soil contamination intensified by the significant outdoor activity inherent with a farming lifestyle (Figure 3.9-7).

The biosphere conceptual model is restricted to the parts of the biosphere that directly contribute to ways that radionuclides could affect a human receptor. The parts of the biosphere considered include the soil, the atmosphere, and flora and fauna. The soil is considered down to the lower bounds of the plant root zone. Radionuclides could move into the soil by irrigation (or volcanic eruption—see Section 3.10). Radionuclides could be removed from the soil by being transported to lower depths or by other processes, such as by plants that are subsequently harvested. The soil itself could be removed by erosion. The atmosphere is a repository for dust particles, some of which could be contaminated by radionuclides. The dust could be available for contaminating plants, animals, and a human receptor. Radionuclides could also affect a human receptor through consumption of the plants, animals, or animal products. Also, a human receptor living in an environment contaminated by radionuclides could be directly irradiated.

Pathways–Once the characteristics of the receptor and environment are known, the biosphere model tracks the pathways by which radionuclides could travel—from the geosphere through the biosphere to the human receptor (e.g., from well water to soil via irrigation, from soil to dust via resuspension, from dust to human lungs via inhalation). There are three exposure pathways by which the human receptors can receive radiation dose: ingestion, inhalation, and external exposure (Figure 3.9-8).

Primary ingestion subpathways include the consumption of drinking water, the consumption of locally produced crops irrigated with contaminated water, and the consumption of meat and dairy products from livestock given contaminated water and fodder. Another ingestion subpathway is the inadvertent ingestion of contaminated soil (e.g., while eating vegetables). The inhalation pathway involves breathing contaminated dust. Important to the inhalation pathway is the amount of time a human receptor spends outdoors. The external-exposure pathway results from proximity to a radiation source that is external to the body. The only external-exposure pathway considered in the biosphere model is exposure to radiation in contaminated soil.

Water Usage–The receptor is modeled as resident in a hypothetical farming community of between 15 to 25 farms supporting about 100 people. The radionuclide concentration in the groundwater is estimated by assuming that all the radionuclides transported across the 20 km boundary in a year are uniformly distributed in the annual quantity of groundwater used by this community—the water usage. This approach eliminates speculation regarding the relative future location of the contamination plume and the individual wells in the proposed community.

The annual water usage by this community is determined from present-day (1997 data) usage, as reported by the State of Nevada. These data allow a statistical estimate of the mean and standard deviation of water usage by the Amargosa Valley residents who could be considered to reside on properties that are used for agricultural activities. Using statistically defined bounds (95-percent confidence limits) for the estimated mean groundwater usage allows a stochastic algorithm to be defined to predict the range of anticipated groundwater usage by the 15 to 25 farms (CRWMS M&O 2000 [151615], Section 3.4).

The water-usage data for Amargosa Valley illustrate that irrigation is the significant consumer of groundwater. Domestic water usage is small in comparison and is ignored in estimating the total water usage (domestic water usage is considered, however, in the modeling of exposure pathways). This approach only impacts, on average, the quantity of groundwater used by about one percent. Figure 3.9-9 shows the water-usage ranges used in the TSPA-SR calculations.

Buildup of Radionuclides in the Soil–To account for the radionuclide buildup in the soil, BDCFs are calculated for each of six periods of cumulative years of irrigation with contaminated groundwater (CRWMS M&O 2000 [151615], Table 3-6, Section 3.2.4.1.2). The periods of previous irrigation are correlated with the period of time it takes until the equilibrium radionuclide concentration in the soil is reached under the continuous irrigation conditions. The first of the six BDCFs are always calculated under the assumption of no prior irrigation (i.e., no radionuclide contamination in soils). The remaining five irrigation periods were selected so that the BDCFs at each period would be approximately equally spaced between their no-prior irrigation values and their long-term asymptotic (leveling off) contamination levels. The radioactive-decay rate, the leaching rate, and the soil-erosion rate are used in the determination of the prior build-up. A major portion of the data related to the modeled soil layer was obtained from a U.S. Department of Agriculture Natural Resource Conservation Service database (CRWMS M&O 2000 [136281], Section 4.1.1) that provides chemical and physical properties for the soils for southern Nye County, including the Amargosa Valley. Figure 3.9-10 presents the processes considered in the calculation of radionuclide buildup in soil.

Not every pathway component is influenced by the changing radionuclide concentration in the soil. For example, the contributions to BDCFs from ingestion of drinking water and the intake of the radionuclide that enters the food chain by deposition on the plant's surfaces during irrigation with contaminated water, are unaffected by radionuclide buildup in the soil. Examples of pathways that are sensitive to radionuclide buildup in soil include external exposure to radiation from contaminated soil, inhalation of resuspended soil particles, and radionuclide uptake by edible plants through their roots.

3.9.2.2 Computer Implementation

The computer program chosen to implement the biosphere conceptual model is GENII-S, a program for statistical and deterministic simulations of radiation doses to humans from radionuclides in the environment (Leigh et al. 1993 [100464]). GENII-S has been accepted by regulatory agencies, including the NRC and the EPA, for the purpose of environmental dose assessment (CRWMS M&O 2000 [151615], Section 3.2.1.3). GENII-S is flexible enough to address the FEPs applicable to Yucca Mountain. Using a comprehensive set of environmental pathway models, the GENII-S program calculates the environmental transport of radionuclides for both of the scenarios considered in TSPA-SR: the use of contaminated groundwater or the use of contaminated soil, resulting from the deposition of volcanic ash containing radionuclides (Section 3.10). Based on the defined source term and exposure scenario, human uptake and exposure to key radionuclides are assessed, and radionuclide media concentration and intake rates are subsequently converted to radiation doses. The output from GENII-S is the set of BDCFs used by the biosphere component of TSPA-SR.

GENII-S has also been subjected to the DOE's software qualification process (CRWMS M&O 1998 [107723]). The qualification process makes use of test cases supplied by the software developer to verify that the software, as installed on project computers, produces outputs that are consistent with values expected for a prescribed set of inputs. Additionally, a special test case, tailored to exercise all the GENII-S pathways and features relevant to Yucca Mountain analyses, has been executed. The expected results of the analysis were calculated by hand, using the equations from the GENII-S mathematical model. Agreement of the GENII-S results and hand calculations were found to be within ±5 percent, and the code was subsequently designated as qualified software. An analysis has also been conducted to compare the BDCFs for the Yucca Mountain scenarios with other BDCF calculations using GENII-S and its predecessor, GENII. The results of this analysis showed a high degree of consistency.

GENII-S and the implemented biosphere model have been validated for use by the YMP. (Validation is a process used to establish confidence that a model adequately represents the phenomenon, process, or system under consideration.) As part of the validation process, an independent technical review concluded that the methods, references, and data sources used were sound and that the GENII-S input values were reasonable for the environment conditions of the biosphere model (CRWMS M&O 2000 [151615], Section 3.2.3).

3.9.2.3 Treatment of Uncertainty and Variability

Because the biosphere system is complex in nature, and much of the general characterization is prescribed by regulation, any biosphere model is a simplified version of the reality on which it is based. To represent uncertainty in some of the model input parameters, the parameters are represented by probability density functions. BDCFs are calculated with GENII-S in a series of probabilistic runs using Latin Hypercube sampling of the probability density functions to determine the set of parameter values used in each run. Thus, uncertainties in the input parameters become propagated in the output BDCF distributions. This technique is known as the Monte Carlo method, and it is the same technique used in the TSPA model to handle uncertainties. (The Latin Hypercube sampling technique is also used to handle uncertainties in the TSPA model.) An individual calculation in the series of runs is called a realization. For the biosphere modeling, 130 realizations are calculated for each radionuclide, and the results—the BDCFs—are 130 equally probable outcomes that are combined to form a probability distribution. Variability is handled in the same manner.

Not all parameters are defined by probability distributions. Parameters are defined by fixed values, if they are well-known or can be shown to be relatively unimportant to the biosphere modeling (e.g., feed storage times). Also, because the receptor is prescribed by regulation to be the average member of the critical group, the lifestyle and consumption parameters are defined as fixed values by taking the average of the probability distributions that were constructed from the regional-survey results. (The original probability distributions for these parameters are used in the parameter sensitivity study, because fixed values do not contribute to the variance in the results [Section 3.9.2.5].) If there is some knowledge of the uncertainty or variability in the parameter values, and if there is indication that the parameters are important to the results, the parameters are defined by probability distributions. In general, for both fixed and distributed parameters, the assessment philosophy is to use generally conservative assumptions to ensure that the results are unlikely to underestimate the corresponding values of BDCFs for the radionuclide transport and uptake conditions and mechanisms considered.

The parameters used in the biosphere model can be classified in two ways: as parameters that influence, or are related to, the transport through (and accumulation of) radionuclides in the biosphere; or as parameters related to characteristics of the human receptor (i.e., consumption patterns, lifestyle characteristics, and land use). Some modeling-input parameters were obtained through field observations and the regional survey, while others were derived from other published sources (see CRWMS M&O 2000 [151615], Section 3.2.4). The GENII-S computer model uses almost 300 input parameters. Many of these parameters are control parameters, many are not applicable to the Yucca Mountain scenarios, and many can only be defined in GENII-S by fixed values. For the TSPA-SR biosphere modeling, 28 parameters are defined by probability distributions. Table 3.9-1 presents a selection of parameters important to the biosphere model and their values.

Table 3.9-1. Selected Input Parameters Used in Biosphere Modeling

Parameter	Distribution Type[a]	Fixed Value for BDCF Calc.
Soil-to-Plant Transfer Scale Factor	Log normal (0.0275,36.4)	—
Animal Uptake Scale Factor	Log normal (0.117,8.51)	—
Inhalation Exposure (hr/yr)	Triangular (3483.38,3918.5,6353.5)	3918.5
Inhalation Exposure Mass Load (g/m^3)	Log normal (7.4E-7,6.4E-5)	—
Breathing Rate (m^3/day)	Constant (23.)	—
Soil Ingestion Rate (mg/day)	Constant (50.)	—
Home Irrigation Rate (in/yr)	Uniform (52.,97.)	—
Crop Resuspension Factor (/m)	Log normal (9.6E-12,7.2E-10)	—
Crop Interception Fraction (-)	Normal (0.044,,0.474)	—
Drinking Water Consumption (L/yr)	Uniform (0.,1500.)	752.8
Leafy Vegetable Grow Time (days)	Triangular (45,64.5,75.)	—
Leafy Vegetable Irrigation Rate (in/yr)	Triangular (28.17,42.11,80.37)	—
Leafy Vegetable Irrigation Time (mo./yr)	Triangular (2.0,3.2,4.9)	—
Leafy Vegetable Yield (kg/m^2)	Triangular (0.59,1.82,4.11)	—
Leafy Vegetable Consumption Rate (kg/yr)	Log uniform (1.2,60.)	15.14
Root Vegetable Consumption Rate (kg/yr)	Log uniform (0.65,30.)	7.81
Fruit Consumption Rate (kg/yr)	Log uniform (0.18,98.)	15.57
Grain Consumption Rate (kg/yr)	Log uniform (8.6E-11,12.)	0.48
Beef Consumption Rate (kg/yr)	Log uniform (7.1E-7,53.)	2.93
Fish Consumption Rate (kg/yr)	Log uniform (6.6E-8,8.8)	0.47
Poultry Consumption Rate (kg/yr)	Log uniform (2.1E-5,11.)	0.8
Milk Consumption Rate (L/yr)	Log uniform (3.E-9,100.)	4.14
Egg Consumption Rate (kg/yr)	Log uniform (0.23,33.)	6.68

Source: CRWMS M&O 2000 [136285], Table 1; CRWMS M&O 2000 [151615] , Tables 3-4, 3-5, 3-16, Sections 3.2.4.1.2, 3.2.4.1.3, 3.2.4.1.4, 3.2.4.1.6, 3.2.4.2.1, 3.2.4.2.

NOTE: [a] The distribution types are parameterized as follows: constant (fixed value), uniform (min, max), triangular (min, mode, max), log uniform (min, max), normal (0.1 percentile, 99.9 percentile), log normal (0.1 percentile, 99.9 percentile). The normal and log-normal distributions are not defined by the typical mean and standard deviation.

No alternative conceptual models or major opposing views to the overall biosphere modeling process are considered in TSPA-SR. The reason for considering only the reference biosphere described here is because it is for the most part prescribed by proposed 10 CFR Part 63 (64 FR 8640 [101680], Section 115). In addition, the biosphere model for the TSPA-SR is consistent with modeling activities being pursued by the international scientific community (BIOMOVS II 1994 [100361]; BIOMOVS II 1996 [100363]; National Research Council 1995 [100018]).

3.9.2.4 Integration of the Biosphere into Total System Performance Assessment

The biosphere is incorporated into the TSPA calculations by the following methodology. The first step of the process involves the calculation of the BDCF distributions, which represent radionuclide-dependent, receptor-dependent doses per unit activity concentration in groundwater (or volcanic ash; see Section 3.10) introduced into the biosphere. The second step of the process involves the determination of the water-usage volume by the hypothetical farming community that contains the critical group and the receptor. The BDCF distributions and the water-usage-volume distribution are input parameters to the TSPA model; at each TSPA realization the BDCF and water-usage-volume values are sampled from the distributions. The third step is within the TSPA model and involves the calculation of the amount of each radionuclide reaching the geosphere/biosphere interface in a given year. The fourth step involves converting the amount of each radionuclide into a concentration, by dissolving the entire amount into the water-usage volume. The fifth step is the calculation of the annual dose incurred by the receptor.

Details of the steps that occur within the TSPA model are as follows. Radionuclide amounts in groundwater are specified in terms of mass flux (specified in units of grams per year [g/yr]) which, when multiplied by the activity for the particular radionuclide (in units of curies per gram [Ci/g]), is converted to an activity flux. When divided by the water-usage volume of the hypothetical farming community (specified in units of liters per year [L/yr]), the activity flux is converted to an activity concentration, specified in units of picocuries per liter (pCi/L). The BDCFs, expressed as annual dose per unit activity concentration in groundwater, are calculated in units of rem/yr per pCi/L. Thus, the radionuclide concentration of a specific radioactive isotope in the water-usage volume is multiplied by the appropriate BDCF to determine the annual radiation dose (in units of rem/yr).

(The annual dose is actually the TEDE received in a single year by the average member of the critical group only as a result of radioactive materials released from the potential geological repository. The TEDE is the sum of the deep-dose equivalent, for external exposures, and the committed effective dose equivalent, for internal exposures. The deep-dose equivalent is the dose equivalent at a tissue depth of 1 cm. The committed effective dose equivalent is the sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent—the product of the absorbed dose in tissue, the quality factor, and all other necessary modifying factors at organs or tissues of reference, that will be received from an intake of radioactive material by an individual during the 50-year period following the intake—to these organs or tissues. This quantity is called the annual dose here for brevity.)

Annual doses are calculated in the TSPA calculation for all radionuclides under consideration. The BDCFs for all the radionuclides are completely correlated in the TSPA model; i.e., if a large BDCF is sampled for one radionuclide, then large BDCFs are sampled for all radionuclides. The sum of the annual doses for all radionuclides is the total annual dose from radionuclide intake and external exposure to radionuclides in the environment for that year. The total annual dose at a given year, averaged for that year over all TSPA realizations, is the mean annual dose for that year. The end product of TSPA-SR is the mean annual doses calculated over time periods of interest to give a mean-annual-dose time history (Section 4).

3.9.2.5 Biosphere Model Results and Their Interpretation

The principal results of the biosphere modeling are BDCFs, which represent doses received annually by the receptor (allowing for the additional accrued dose as a result of the time each radionuclide spends within the receptor) for each unit of radionuclide concentration in groundwater (or volcanic ash; Section 3.10) introduced into the biosphere. For each radionuclide, GENII-S was used to perform 130 realizations of the biosphere model to generate a set of BDCFs. These results were fitted with a log-normal or shifted log-normal distribution, and a statistical test was used to verify the goodness of fit. In the TSPA calculations, the BDCF values were sampled from these log-normal or shifted log-normal distributions for each radionuclide for each TSPA realization.

The stochastic biosphere modeling shows that for a given radionuclide, the BDCF varies only in a tight range about the mean value, primarily because the consumption parameters are treated as fixed values to model an average member of the critical group. Figure 3.9-11 presents the histogram and log-normal distribution fitted to the 130 BDCFs calculated by GENII-S for ^{237}Np. BDCF distributions have been developed for 18 radionuclides. The radionuclide were selected through an analysis designed to determine which radionuclides should be included in the TSPA-SR, based on their potential contribution to dose (Section 3.5.1; 16 radionuclides for the nominal scenario plus two radionuclides—^{90}Sr and ^{137}Cs—for the Human Intrusion analysis). Table 3.9-2 shows the BDCFs for the 18 radionuclides and the distribution parameters used in the TSPA model.

Table 3.9-2. Biosphere Dose Conversion Factors and Soil Buildup Factors for Radionuclides Introduced into the Biosphere through Irrigation with Contaminated Groundwater

Radionuclide	BDCF (mrem/yr per pCi/L)			Soil Buildup Factor
	Geometric Mean	Geometric Standard Deviation	Offset	
^{14}C	0.5536E-03	1.5177	3.4675E-03	1.00
^{90}Sr	1.121E-01	2.736	1.525E-01	1.93
^{99}Tc	1.4948E-03	1.8423	2.1631E-03	1.01
^{129}I	3.562E-01	1.187	—	1.00
^{137}Cs	1.841E-01	1.163	—	2.21
^{227}Ac	1.801E+01	1.162	—	1.01
^{229}Th	5.392E+00	1.167	—	2.85
^{232}U	2.064E+00	1.150	—	1.13
^{233}U	3.848E-01	1.161	—	1.03
^{234}U	3.769E-01	1.162	—	1.03
^{236}U	3.564E-01	1.164	—	1.03
^{238}U	3.512E-01	1.159	—	1.04
^{237}Np	6.738E+00	1.163	—	1.01
^{238}Pu	4.109E+00	1.161	—	1.01

Table 3.9-2. Biosphere Dose Conversion Factors and Soil Buildup Factors for Radionuclides Introduced into the Biosphere through Irrigation with Contaminated Groundwater (Continued)

Radionuclide	BDCF (mrem/yr per pCi/L)			Soil Buildup Factor
	Geometric Mean	Geometric Standard Deviation	Offset	
^{239}Pu	4.976E+00	1.151	—	1.10
^{240}Pu	4.953E+00	1.151	—	1.10
^{241}Am	5.012E+00	1.156	—	1.08
^{243}Am	5.030E+00	1.163	—	1.62

Source: CRWMS M&O 2000 [151615], Tables 3-19 to 3-20

NOTE: BDCF = biosphere conversion factor

Table 3.9-2 also presents soil buildup factors calculated for each radionuclide. Most of the BDCFs presented in Table 3.9-2 increased with the duration of previous irrigation (the readily leached, nonsorbing radionuclides, e.g., ^{14}C and ^{129}I, showed little or no increase) (CRWMS M&O 2000 [151615], Table 3-17, Section 3.3.1.1.1), reflecting radionuclide buildup in the soil. For most radionuclides, however, the increase in dose because of soil buildup is less than 15 percent. For these radionuclides, the conservative BDCF distributions appropriate to the longest periods of irrigation are used in TSPA-SR. Five radionuclides, ^{90}Sr, ^{137}Cs, ^{229}Th, ^{232}U and ^{243}Am, display greater than 15 percent increase in dose because of soil buildup. For ^{90}Sr, ^{137}Cs, and ^{232}U, the time to approach the maximum buildup is within a few hundred years; for ^{229}Th and ^{243}Am, the time to approach the buildup limit is a few thousand years. Soil loss thus has a greater effect on dose for ^{229}Th and ^{243}Am than for ^{90}Sr, ^{137}Cs, and ^{232}U. The estimated annual rate of soil loss for the major soil series present in the 5-km area surrounding the proposed location of the critical group (junction of U.S. 95 and Nevada State Route 373) is generally between 0.06 and 0.08 cm/yr (CRWMS M&O 2000 [136281], Table 4, p. 16), implying that a 15-cm soil layer would be eroded in less than 250 yr. Once soil loss is considered, the maximum buildup factor for ^{229}Th and ^{243}Am is relatively small—less than 20 percent. For TSPA-SR, the conservative approach of using the asymptotic (i.e., long-time irrigation buildup period) BDCF mean is used.

Sensitivity and Uncertainty Analyses--Sensitivity analyses have been conducted to determine the factors that influence the BDCFs for the contaminated-groundwater-use scenario (CRWMS M&O 2000 [151615], Section 3.3.1.2). Two aspects of the biosphere modeling have been investigated: the pathways that contribute most of the BDCF, and the uncertain or variable parameters that contribute most of the variance in the BDCF distributions.

Analysis of the contaminated-groundwater-use results of the biosphere modeling shows that ingestion exposure accounts for essentially all of the magnitude of the BDCFs. For most radionuclides, the most important pathway within ingestion exposure is the drinking-water-ingestion pathway, followed by the leafy-vegetables pathway. For most radionuclides, all other pathways generally affect the BDCFs at a relatively insignificant level. For the ^{14}C BDCF, consumption of fish raised in ponds filled from groundwater sources is the greatest contributor (more than 90 percent). Fish consumption is also the leading contributor to the BDCF for ^{137}Cs, followed by drinking water, leafy vegetables, and meat.

To determine the parameters to which the dose distributions are most sensitive, rank regression is employed to assess the relationship between the model input and the output. The standard regression coefficient is the metric used to judge the contribution of the variance in the parameter to the variance in the dose (i.e., how much the spread in the distribution of the parameter causes spread in the distribution of the dose). Although many of the receptor parameters used to calculate the BDCF distributions are set to fixed values because of the need to model an average member of the critical group, for the sensitivity study, the distributions of these parameters are used. These parameters include the consumption rates for drinking water and foodstuffs. Forty independent variables (defined by probability distributions) are evaluated in the sensitivity study for 18 different radionuclides. (Twenty-eight stochastic variables are used to calculate the BDCFs.)

The results of the regression analysis show that the leafy-vegetable consumption rate (a stochastic parameter in the sensitivity studies, but a fixed parameter in the BDCF calculations) is the most significant contributor to variance in the dose for all radionuclides, except ^{14}C, ^{99}Tc, and ^{137}Cs. For ^{99}Tc, the leafy-vegetable consumption rate is the second-most contributor. The drinking-water consumption rate is the second leading contributor to variance for all of the radionuclides, except ^{14}C, ^{99}Tc, and ^{137}Cs. Crop-interception fraction (the fraction of contamination from irrigation that is intercepted by and adheres to the plant surface) is the third-leading contributor to variance, except, again, for ^{14}C, ^{99}Tc, and ^{137}Cs. For ^{14}C and ^{137}Cs, variance in fish consumption rate accounts for most of the variance in dose. For ^{99}Tc, most of the variance in dose comes from the milk consumption rate. Of the parameters that are actually treated stochastically in the calculation of the BDCFs, the ones that are important to variance are the crop-interception fraction and the soil-plant transfer scaling factor (a factor adjusting the amount of radionuclides taken up by plants).

3.10 VOLCANISM

As described in Section 2.1, igneous activity has been identified as a disruptive event that has a potential to affect long-term performance of the potential repository. Yucca Mountain is in a region that has had repeated volcanic activity in the geologic past, and the site-specific analysis summarized in the following sections indicates that, although the probability of recurrence at Yucca Mountain is small during the next 10,000 years, it is greater than the one chance in 10,000 in 10,000 years probability criterion defined by the NRC in proposed 10 CFR 63.114(d) (64 FR 8640 [101680]). Volcanic activity therefore cannot be excluded from the TSPA-SR on the basis of low probability. The following sections describe the likelihood and characteristics of igneous activity at Yucca Mountain and the scenarios selected for quantitative analysis in the TSPA.

If igneous activity occurs at Yucca Mountain, possible effects on the potential repository can be grouped into three broad areas depending on the nature of the igneous event.

- Does igneous activity occur at Yucca Mountain without directly intersecting the potential repository? Igneous events that do not intersect the potential repository are examined in the context of FEPs related to the indirect effects of intrusion. As discussed in Section 3.10.2.1, the FEPs associated with igneous intrusions that do not intersect the potential repository have been shown to have insignificant consequences on expected annual dose, and are not included in the TSPA model.

- Does a volcanic eruption occur at the potential repository? Volcanic eruptions within the potential repository could entrain waste within rising magma and pyroclastic material, bringing waste to the surface. As discussed in Section 3.10.2.2, a volcanic eruption scenario is included explicitly in the TSPA, with simulation of atmospheric transport of volcanic ash contaminated with radioactive waste and subsequent human exposure downwind.

- Does an igneous intrusion intersect the potential repository? Essentially all circumstances under which an eruption might occur at the potential repository would also include the intrusion of magma or pyroclastic material into the potential repository. An intrusive dike could intersect the potential repository without resulting in an eruption directly at the potential repository location, however. Regardless of whether or not an eruption occurs, an intrusion could damage waste packages and expose waste to groundwater. As discussed in Section 3.10.2.3, an igneous intrusion groundwater transport scenario is explicitly included in the TSPA, with simulation of radionuclide transport away from damaged packages and subsequent human exposure from contaminated groundwater.

The volcanic eruption and igneous intrusion groundwater transport scenarios together form the Igneous Activity Scenario Class (see Section 2.1 for a discussion of scenario classes), shown schematically in Figure 3.10-1. Because volcanic disruption of the potential repository has the potential to cause major changes in the behavior of the disposal system, the TSPA wheel used in previous sections of this chapter to describe the linkage between the main model components needs extensive modification for igneous activity.

Figure 3.10-2 shows the model components for the volcanic eruption scenario. Each of these components is described in more detail in Section 3.10.2.2. The scenario begins with an eruptive event, which is characterized in the TSPA by both its probability and its physical properties such as energy and volume of the eruption, composition of the magma, and properties of the pyroclastic ash. Interactions of the eruption with the potential repository are described in terms of the damage to the EBS and the waste package. Characteristics of the waste form in the eruptive environment are described in terms of waste particle size. Atmospheric transport of waste in the volcanic ash plume begins with entrainment of waste particles in the pyroclastic eruption and is affected by wind speed and direction. BDCFs are developed specifically for exposure pathways relevant to atmospheric deposition of contaminated ash, rather than for the groundwater pathways considered for nominal performance, as described in Section 3.9. As a final step, the volcanic eruption BDCFs are used to determine radiation doses resulting from exposure to contaminated volcanic ash 20 km from the potential repository.

Figure 3.10-3 shows the model components for the igneous intrusion groundwater transport scenario. As described in more detail in Section 3.10.2.3, many model components in this scenario are essentially unchanged from the nominal scenario. The scenario begins with an intrusive event, which is characterized in the TSPA by its probability and physical properties. Although the intrusion damages waste packages and other components of the EBS, FEPs analyses have concluded that it does not significantly alter the long-term flow of water through the mountain (CRWMS M&O 2000 [146681], Section 6.2.16), and the scenario, therefore, uses nominal scenario models to describe groundwater flow and radionuclide transport through the mountain. Treatment of the biosphere is unchanged from the nominal scenario, and the BDCFs are those developed for exposure pathways related to contaminated groundwater.

The remainder of Section 3.10 is organized into three main topics. Section 3.10.1 describes the construction of the TSPA-SR conceptual model for igneous activity in the Yucca Mountain region, summarizing the history of volcanic activity in the region and the likelihood and nature of possible future activity. Section 3.10.2 describes the implementation of the model in the TSPA. Specifically, Section 3.10.2 identifies the important FEPs that are potentially relevant to igneous activity at Yucca Mountain and describes the conceptual and computational models for volcanic eruption and igneous intrusion groundwater transport implemented in performance assessment. Section 3.10.3 provides interpretation of selected results from the TSPA-SR models for igneous disruption.

3.10.1 The Conceptual Model for Igneous Activity at Yucca Mountain

The conceptual model for igneous activity at Yucca Mountain provides the basis for the characterizations of uncertainty in the probability of igneous disruption and its consequences that are required by the TSPA-SR. There are three main components of the conceptual model: a review of the history of past igneous activity in the Yucca Mountain region; development of an estimate of the likelihood of future igneous activity at the potential repository site; and an analysis of the possible characteristics of a future eruption at the site. Each of these components is based on observations of the past geologic record and, for the characteristics of an eruption, observations of modern analogs. Basing the conceptual model for possible future igneous activity on the past record and modern analogs is consistent with the proposed regulatory requirement to assume that the "[e]volution of the geologic setting shall be consistent with present knowledge of natural processes" (proposed 10 CRF 63.115(a)(4), [64 FR 8640 [101680], p.8677]).

Discussions below of the volcanic history of the Yucca Mountain region and the probability of future igneous activity are taken from *Characterize Framework for Igneous Activity at Yucca Mountain, Nevada (T0015)* (CRWMS M&O 2000 [141044]). The discussion of the characteristics of an eruption is based on *Characterize Eruptive Processes at Yucca Mountain, Nevada* (CRWMS M&O 2000 [142657]) and *Dike Propagation Near Drifts* (CRWMS M&O 2000 [142635]). Additional detail on each topic is available in these AMRs, which were developed as part of a set of analyses supporting the *Disruptive Events Process Model Report* (CRWMS M&O 2000 [141733]).

3.10.1.1 Volcanic History of the Yucca Mountain Region

Two major types of volcanism have occurred in the Yucca Mountain region: an early phase of Miocene (approximately 24 to 5 million years ago) silicic volcanism that is not expected to recur; and, a more recent phase of Miocene and post-Miocene basaltic volcanism that is the basis for the TSPA-SR analysis of igneous disruption).

Silicic volcanism in the region occurred between 15 and 7.5 million years ago, forming the major calderas and ash-flow tuffs of the southwestern Nevada volcanic field (Sawyer et al. 1994 [100075]). Silicic volcanism was approximately coincident with a major period of extensional tectonics that occurred primarily between 13 and 9 million years ago (Sawyer et al. 1994 [100075], Figure 4). In terms of eruption volume, magmatic activity of all types peaked in the region between 13 and 11 million years ago, with the eruption of over 5,000 km^3 of ash-flow tuffs, and has been in decline since. Yucca Mountain itself is an uplifted, erosional remnant of a tuff deposit formed during this early phase of volcanic activity.

Basaltic volcanism began, as extension rates waned, about 11 million years ago (CRWMS M&O 1998 [100129], Figure 3.9-2), during the latter part of the caldera-forming phase, and small-volume basaltic volcanism has continued into the Quaternary (the last 1.6 million years). Approximately 99.9 percent of the volume of the southwestern Nevada volcanic field had erupted by about 7.5 million years ago, when the Stonewall Mountain volcanic center was active as the last silicic caldera system of the field. The last 0.1 percent of eruptive volume of the southwestern Nevada volcanic field consists entirely of basalt erupted since 7.5 million years ago (CRWMS M&O 1998 [100129], Figure 3.9-5). Based on total eruption volume, the southwestern Nevada volcanic field has virtually ceased eruptive activity since about 7.5 million years ago. Although basaltic volcanic activity has continued in the Quaternary, the Yucca Mountain region is reasonably characterized as one of the least active basaltic volcanic fields in the western United States (e.g., CRWMS M&O 1998 [106491], Figure 4-2), for post-Miocene basalts of Crater Flat.

Post-caldera basalts in the Yucca Mountain region can be divided into two episodes: Miocene (eruptions between approximately 9 and 7.3 million years ago) and post-Miocene (eruptions between approximately 4.8 and 0.08 million years ago). The time interval of about 2.5 million years between these episodes is the longest eruptive hiatus of basalt in the Yucca Mountain region during the last 9 million years (CRWMS M&O 1998 [135988], Table 3.1). This eruptive hiatus also marks a distinct shift in the locus of post-caldera basaltic volcanism in the Yucca Mountain region to the southwest (CRWMS M&O 1998 [100129], Figure 3.9-6). The Miocene basalts and post-Miocene basalts are, thus, both temporally and spatially distinct. This observation emphasizes the importance of considering the age and location of the post-Miocene basalts (approximately the past 5 million years of the volcanic history of the Yucca Mountain region) when calculating the volcanic hazard to the potential Yucca Mountain repository.

The post-Miocene basalts formed during at least six episodes of volcanism (based on age groupings) that occurred within 50 km of the potential repository (Figure 3.10-4). These six episodes, in order of decreasing age, consist of: (1) Thirsty Mesa, (2) Pliocene Crater Flat and Amargosa Valley, (3) Buckboard Mesa, (4) Quaternary Crater Flat, (5) Hidden Cone and Little Black Peak (the Sleeping Butte centers), and (6) Lathrop Wells. Three basalt episodes are in or near the Crater Flat topographic basin, within 20 km of Yucca Mountain. The total eruption volume of the post-Miocene basalts is about 6 km^3. The volume of individual episodes has decreased progressively through time, with the three Pliocene (approximately 5 to 1.6 million years ago) episodes having volumes of approximately 1 to 3 km^3 each and the three Quaternary episodes having a total volume of less than 0.5 km^3 (CRWMS M&O 1998 [100129], Figure 3.9-2; Table 3). All of the Quaternary volcanoes are similar in that they are of small volume (approximately 0.1 km^3 or less, Table 3.10-1), and typically consist of a single main scoria cone surrounded by a small field of *aa* basalt flows, which commonly extend approximately 1 km from the scoria cone.

Table 3.10-1. Estimated Volume and $^{40}Ar/^{39}Ar$ Age[1] of Quaternary Volcanoes in the Yucca Mountain Region

Volcano	Volume (km^3)[2]	Volume (km^3)[3]	Age (m.y.)[5]
Makani Cone	0.006		1.16-1.17
Black Cone	0.105	0.07	0.94-1.10
Red Cone	0.105		0.92-1.08
Little Cones	0.002	>0.01[4]	0.77-1.02
Hidden Cone	0.03		0.32-0.56
Little Black Peak	0.03		0.36-0.39
Lathrop Wells Cone	0.14		0.074-0.084

Source: CRWMS M&O 2000 [141044]

NOTES: [1] $^{40}Ar/^{39}Ar$ dates provide the most complete and self-consistent chronology data set for Quaternary volcanoes of the Yucca Mountain region. A full discussion of other chronology methods used to date basaltic rocks in the Yucca Mountain region can be found in *Synthesis of Volcanism Studies for the Yucca Mountain Site Characterization Project* (CRWMS M&O 1998 [105347]). Other chronology methods may not provide consistent or accurate estimates of the time of eruption.

[2] CRWMS M&O 1998 [135988], Table 3.1; DTN: LA0004FP831811.002 [149593]

[3] Stamatakos et al. 1997 [138819], p. 327

[4] Accounts for volume of buried flows detected by ground magnetic surveys

[5] Range of ages from *Synthesis of Volcanism Studies for the Yucca Mountain Site Characterization Project* (CRWMS M&O 1998 [105347], Table 2.B) and Heizler et al. (1999 [107255], Table 3) (DTN: LAFP831811AQ97.001 [144279]).

The seven (or eight, if Little Cones is counted as two) volcanoes associated with the three Quaternary volcanic episodes occur to the south, west, and northwest of Yucca Mountain in a roughly linear zone defined as the Crater Flat Volcanic Zone (Crowe and Perry 1990 [100973], p. 328). Five of seven Quaternary volcanoes are in or near Crater Flat and lie within 20 km of the Yucca Mountain Site (Figure 3.10-4). Models that relate volcanism and structural features in the Yucca Mountain region have emphasized the Crater Flat basin because of the frequency of volcanic activity associated with Crater Flat and its proximity to the potential Yucca Mountain repository (e.g., Smith et al. 1990 [101019], p. 84; Connor and Hill 1995 [102646], p. 10,122).

3.10.1.2 Estimating the Probability of Future Igneous Activity in the Yucca Mountain Region

The probability of future igneous activity in the Yucca Mountain region that is used in the TSPA-SR is based on the Probabilistic Volcanic Hazard Analysis conducted by the DOE in 1995 and 1996 (CRWMS M&O 1996 [100116]). Ten experts in the field of volcanology evaluated available data on past volcanic activity in the region and provided expert judgement on the probability of future igneous activity. Their judgments (elicitations) were then combined to produce an integrated assessment of the volcanic hazard that reflects a range of alternative scientific interpretations. Details of the identification of the experts, presentation of available data to them, and the elicitation process are available in the summary report of the probabilistic volcanic hazard analysis (CRWMS M&O 1996 [100116]).

The probabilistic volcanic hazard analysis focused on the volcanic hazard at the site expressed as the probability of intersection of the potential repository by an intrusive basaltic dike, rather than on the probability of a volcanic eruption. Each of the 10 experts independently arrived at a probability distribution for the annual frequency of intersection of the repository footprint by a dike that typically spanned roughly 2 orders of magnitude (CRWMS M&O 1996 [100116], Figure 4-31). From these individual probability distributions, an aggregate probability distribution for the annual frequency of intersection of the potential repository footprint by a dike was computed that reflected the uncertainty across the entire expert panel (CRWMS M&O 1996 [100116], Figure 4-32). The individual expert's distributions were combined using equal weights to obtain the aggregate probability distribution. The mean value of the aggregate probability distribution defined by the probabilistic volcanic hazard analysis was 1.5×10^{-8} dike intersections per year, with a 90 percent confidence interval of 5.4×10^{-10} to 4.9×10^{-8} (CRWMS M&O 1996 [100116], p. 4-10). These probabilities have been recalculated for the TSPA-SR to account for the current repository footprint and probabilistic volcanic hazard analysis results have been further interpreted to yield the probability of a volcanic eruption within the potential repository footprint, conditional on the occurrence of an intrusive dike (CRWMS M&O 2000 [141044]). Probability values for the current potential repository footprint are summarized in Table 3.10-2. Probabilities used in the TSPA-SR are sampled from the distribution calculated for the full potential repository footprint, including both the primary and contingency disposal blocks.

Table 3.10-2. Summary Frequencies of Disruptive Events

Repository Footprint (EDA II)	Hazard Level	Annual Frequency of Intersection of Repository by a Dike	Weighted Conditional Probability of No Eruptive Centers	Annual Frequency of Occurrence of One or More Eruptive Centers within Repository
Primary Block	5th percentile	6.6×10^{-10}	0.58	2.8×10^{-10}
	Mean	**1.4×10^{-8}**	0.53	**6.7×10^{-9}**
	95th percentile	4.7×10^{-8}	0.53	2.2×10^{-8}
Primary + Contingency Blocks	5th percentile	7.6×10^{-10}	0.56	3.3×10^{-10}
	Mean	**1.6×10^{-8}**	0.50	**7.7×10^{-9}**
	95th percentile	5.0×10^{-8}	0.51	2.5×10^{-8}

Source: Output data. DTN: LA0004FP831811.004 [151391]; CRWMS M&O 2000 [141044], Table 13

3.10.1.3 Characteristics of a Hypothetical Future Igneous Event at Yucca Mountain

Models of the consequences of an igneous intrusion into, or a volcanic eruption through, the potential Yucca Mountain repository require specific information about the nature of the igneous event and the response of the potential repository. Direct observations of volcanic processes in the subsurface are extremely rare and there are no precedents for modeling the intrusion of Yucca Mountain-type basaltic magmas into a mined repository. Information used in the TSPA-SR model to characterize the intrusive and eruptive processes comes from three sources: examination of the geologic record of past intrusive and eruptive events in the Yucca Mountain region; observations of eruptive processes during analogous modern volcanic events elsewhere in the world; and consideration of the range of physical processes that might occur during the interaction between the potential repository and an igneous dike or conduit. The first two sources of information are described in *Characterize Eruptive Processes at Yucca Mountain, Nevada* (CRWMS M&O 2000 [142657]), and, to a lesser extent, *Characterize Framework for Igneous Activity at Yucca Mountain (T0015)* (CRWMS M&O 2000 [141044]). Analyses of the interactions between a dike and a drift are described in *Dike Propagation Near Drifts* (CRWMS M&O 2000 [146681]). Additional information regarding the response of the waste package and waste form to the igneous environment is provided in *Miscellaneous Waste-Form FEPs* (CRWMS M&O 2000 [146498]) and the calculation *Waste Package Behavior in Magma* (CRWMS M&O 1999 [121300]).

Specific information developed to support the TSPA-SR conceptual models for igneous disruption of the potential repository includes the following:

- The geometry of an intrusion: dike width, length in the potential repository, azimuth, and the number of dikes that could occur as part of a single intrusive event.

- The geometry of an eruption: conduit diameter at the potential repository depth, and the number of conduits (also called eruptive centers and vents) that intersect drifts and that could be associated with a single intrusive event.

- Physical and chemical properties of the magma: temperature, density, volatile content.

- Intrusive properties: magmatic ascent velocity, fragmentation depth.

- Eruptive properties: pyroclastic ascent velocity, eruption power, eruption duration, eruption volume (mass discharge rate), ash particle size and shape, ash density.

- Dike and potential repository interactions: environmental conditions in the drift, response of the waste package, extent of the magmatic damage in the drifts (including the number of waste packages damaged by both intrusion and eruption), behavior of the waste form in the eruptive environment.

- Atmospheric properties: wind speed and direction, ash dispersion, air density and viscosity.

Uncertainty regarding the estimates is typically included in the TSPA-SR through the specification of distributions of reasonably possible values. Quantitative values for selected parameter distributions used in the TSPA-SR are given in Section 3.10.2, and are not repeated here.

3.10.2 Implementation of the Performance Assessment Model

The TSPA-SR model for igneous disruption of the potential repository has been constructed following evaluation of a comprehensive list of relevant FEPs that have a potential to affect long-term performance. As described in Section 2.1, these FEPs have been screened against regulatory criteria of consequence and probability. Those FEPs that have been found to either have no significant impact on expected annual dose or to occur with a probability below one chance in 10,000 in 10,000 years have been excluded from the quantitative TSPA on the basis of low consequence or low probability. The remaining FEPs relevant to igneous activity have been included in the TSPA-SR through two separate models for igneous disruption: a model for volcanic eruptions that intersect drifts and bring waste to the surface; and a model for igneous intrusions that damage waste packages and expose radionuclides to groundwater transport processes.

Section 3.10.2.1 summarizes the evaluation of the FEPs relevant to igneous activity, based on work documented in *Disruptive Events FEPs* (CRWMS M&O 2000 [146681]). Section 3.10.2.2 describes the TSPA-SR model for volcanic eruption, based on work documented in *Igneous Consequence Modeling for the TSPA-SR* (CRWMS M&O 2000 [139563]) and other supporting analyses. Section 3.10.2.3 describes the TSPA-SR model for possible radionuclide releases resulting from igneous intrusions. As contrasted to a volcanic eruption, in which radioactive waste could be transported directly to the location of the critical group in an ash plume, releases from an intrusion could occur as groundwater transports radionuclides from damaged waste packages. The TSPA-SR model for radionuclide releases occurring by groundwater transport following igneous intrusion is documented in *Igneous Consequence Modeling for the TSPA-SR* (CRWMS M&O 2000 [139563]).

3.10.2.1 Features, Events, and Processes Potentially Relevant to Igneous Activity

The eight Primary FEPs related to igneous activity are listed in Table 3.10-3, together with a statement of their screening status (included in or excluded from the TSPA-SR system-level analysis). For FEPs that are excluded from the TSPA-SR, the table contains a summary statement of the screening criterion used (low probability or low consequence), as described in Section 2.1. Full discussions of the technical bases for the screening decisions and discussions of the more detailed secondary FEPs associated with these FEPs are contained in *Disruptive Events FEPs* (CRWMS M&O 2000 [146681]).

Table 3.10-3. Summary of Primary Igneous FEPs Screening Decisions

YMP FEP Database Number	FEP Name	Screening Decision	Screening Basis
1.2.03.03.00	Seismicity associated with igneous activity	Exclude for indirect effects / Include for drip shield and fuel-rod cladding damage	Low Consequence
1.2.04.01.00	Igneous activity (Note: Also effects on faults, topography, rock stresses, groundwater temperatures & drift integrity)	Include: for direct effects / Exclude: for indirect effects	Low Consequence of Indirect Effects
1.2.04.02.00	Igneous activity causes changes to rock properties	Exclude	Low Consequence
1.2.04.03.00	Igneous intrusion into repository	Include	
1.2.04.04.00	Magma interacts with waste	Include	

Table 3.10-3. Summary of Primary Igneous FEPs Screening Decisions (Continued)

YMP FEP Database Number	FEP Name	Screening Decision	Screening Basis
1.2.04.05.00	Magmatic transport of waste	Exclude for transport in liquid magma and other types of transport / Include for transport through eruptive events	Low Consequence
1.2.04.06.00	Basaltic cinder cone erupts through the repository (Note: Also entraining waste)	Include	
1.2.04.07.00	Ashfall	Include	
1.2.06.00.00	Hydrothermal Activity	Exclude	Low Probability
1.2.10.02.00	Hydrologic response to igneous activity (Note: Includes groundwater flow directions; water level, chemistry, temperature; change in rock properties)	Exclude	Low Consequence

Source: CRWMS M&O 2000 [146681]

As discussed in the following section, FEPs that are listed as "include" have been included in the TSPA models through a variety of approaches. In some cases, such as the FEP "Magma Interacts with Waste," many of the possible types of interactions have been included, through conservative assumptions, about the behavior of the waste packages and waste form in the magmatic environment. For example, although the TSPA-SR does not explicitly model degradation of the waste package in an eruptive conduit, this aspect of the FEP has been included through the bounding assumption that any waste package intersected by an eruptive conduit is sufficiently damaged that it provides no further protection for the waste and that all waste contained within it is available to be entrained in the eruption. Other FEPs, such as "Ashfall," are modeled in a more realistic and explicit manner, as described in Section 3.10.2.2.

The FEPs listed as "exclude" in Table 3.10-3 have been evaluated and shown to have no significant effect on overall performance. Screening arguments are summarized from *Disruptive Events FEPs* (CRWMS M&O 2000 [146681]) and, for hydrothermal activity, from *Features, Events, and Processes in UZ Flow and Transport* (CRWMS M&O 2000 [142945]) in the following paragraphs.

Seismicity Associated with Igneous Activity–Seismicity related to igneous processes, such as basaltic volcanoes and dike injection, has been treated in the TSPA-SR in the same manner as seismicity from all sources. For example, indirect effects of seismicity on groundwater flow are excluded from the TSPA on the basis of low consequence (CRWMS M&O 2000 [142945]). Other effects of seismicity, such as damage to cladding due to ground motion during seismic events, are included in the TSPA (see Section 3.x.x, Waste Form). Screening of all seismic FEPs is based on work done as part of the expert elicitation *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada* (USGS 1998 [100354]). Because the probabilistic seismic hazard analyses considered seismicity related to igneous activity in the development of the probability models for fault displacement and ground motion, screening decisions based on the probabilistic seismic hazard analyses apply to seismicity associated with igneous activity as well as other causes.

Igneous Activity–Direct effects of igneous activity, including intrusion, cinder cone formation, and ashfall, are included in the TSPA as described in the following section. Indirect effects that have been excluded on the basis of low consequence include effects on faults, changes in topography, seismic effects, changes in rock properties, and changes in local and regional hydrology. Seismic and hydrologic aspects of igneous activity and changes in rock properties are addressed as separate FEPs discussed elsewhere in this section and are excluded on the basis of low consequence. Faulting associated with igneous activity was considered in the overall analysis of faulting in the Probabilistic Seismic Hazard Analyses and is excluded on the basis of low consequence for reactivation of existing faults and low probability for the creation of new faults. Changes in topography due to volcanic activity have been excluded on the basis of low consequence because of the relatively small volumes of the cinder cones associated with the type of igneous activity possible in the Yucca Mountain region.

Igneous Activity Causes Changes to Rock Properties–Igneous activity could cause changes in the physical, chemical, and hydrologic properties of rock near intrusive bodies. Such changes are potentially important to performance if the affect groundwater flow and radionuclide transport in large regions of rock. Analog studies at other sites on the Nevada Test Site show, however, that effects of igneous activity, including mineralogic changes, are generally limited to regions within 10 meters of dikes, and that fracturing and jointing in the surrounding rock is generally parallel to the contacts of the dike (CRWMS M&O 1998 [123201], pp. 5-32, 5-41). Because of the limited extent of the effects and because the preferred orientation of dikes in the Yucca Mountain region (NE-SW) is subparallel to the maximum principal transmissivity direction in the SZ of approximately N30E (Ferrill, Winterle et al. 1999 [118941], p. 1) intrusions are not expected to significantly affect groundwater flow patterns. Changes in rock properties due to igneous activity that does not intersect the potential repository are, therefore, excluded from the TSPA on the basis of low consequence.

Magmatic Transport of Waste–Transport of waste in a pyroclastic eruption is included explicitly in the TSPA-SR model. However, transport of waste in liquid magma (e.g., dissolved or entrained in lava that might reach the surface) has been excluded on the basis that surficial lava flows associated with this type of volcanism are of very limited extent and could not reach the location of the critical group. Consequences of such transport are therefore insignificant compared to the consequences of the pyroclastic transport pathway that has been modeled.

Hydrothermal Activity–Hydrothermal activity usually represents the last cooling stage of magmatic intrusion when residual water is passed from the crystallizing melt, and/or its heat flux is sufficient to set up local deep groundwater convection such that hot, mineral laden solutions pass upwards to cool and precipitate in a halo around the intrusion; the country rock literally stews in hot fluid for periods of hundreds to thousands of years. Evidence of past hydrothermal activity associated with Miocene silicic volcanism is conspicuous in the Calico Hills and along the south flank of Shoshone Mountain, but comparable hydrothermal activity has not been identified at Yucca Mountain in association with basaltic activity. Yucca Mountain is located outside the caldera margin (see Figure 3.10-4), and was never sufficiently close to a hydrothermal source. Because silicic volcanism has not occurred in the Yucca Mountain region in approximately 7.5 million years and is not expected to recur in the future, the effects of hydrothermal activity are excluded from TSPA on the basis of low probability.

Hydrologic Response to Igneous Activity–Changes in groundwater flow that might occur in response to igneous activity will be controlled by changes in the hydrologic properties of the rocks near the intrusive dike. As noted above, these changes will be limited in extent and generally parallel to existing trends in hydrologic properties, and thus would have no significant effect on the calculation of the expected annual dose. Hydrologic effects, therefore, are excluded on the basis of low consequence.

3.10.2.2 TSPA-SR Model for Volcanic Eruption

The TSPA model for the consequences of a volcanic eruption at Yucca Mountain is a simplification of the complex processes that could occur during an eruptive event intended to yield a reasonable estimate of the concentrations of radionuclides that could reach the critical group following an eruption. At a conceptual level, the overall model is straightforward (Figure 3.10-5). An igneous dike rises through the earth's crust and intersects one or more drifts in the potential repository. An eruptive conduit forms somewhere along the dike as it nears the land surface, feeding a volcano at the surface. Waste packages in the path of the conduit are sufficiently damaged that they provide no further protection, and the waste is available to be entrained in the eruption. Volcanic ash is contaminated, erupted, and then transported by wind toward the critical group. Ash settles out of the plume as it is transported downwind, resulting in an ash layer on the land surface. Members of the critical group receive a radiation dose from various pathways associated with the contaminated ash layer, including inhalation and ingestion.

Implementation of this model in the TSPA-SR is shown in Figure 3.10-6. Information about eruption characteristics, the probability of eruptive conduits forming within the potential repository, and the potential repository response to eruption are used to develop a distribution of parameter values characterizing uncertainty in the extent of damage to waste packages and the amount of waste available to be entrained in the eruption. Entrainment of waste and atmospheric transport of contaminated ash is modeled using the ASHPLUME code, Version 1.4LV (CRWMS M&O 1999 [150744]), yielding a distribution of results characterizing uncertainty in the concentration of waste particles on the ground surface. BDCFs calculated for the volcanic eruption scenario using the GENII-S code (CRWMS M&O 1998 [107723]) are used to calculate doses, and results are weighted by the eruption probability to determine the probability-weighted dose needed to calculate the expected annual dose required by proposed 10 CFR 63.113(b) (64 FR 8640 [101680]).

Individual components of the model addressing the behavior of the dike and conduit, response of the waste package, the modeling of the ash plume, and the calculation of BDCFs for the volcanic eruption scenario are discussed in the following sections. Process model factors, as introduced in Section 3.1 for the entire modeling system, are summarized for volcanic eruption in Table 3.10-4. TSPA parameter distributions associated with these factors and used in the computational model are also summarized in Table 3.10-4. Figure 3.10-7 shows the key elements of the eruption scenario, major model inputs and outputs, and the major bases for confidence that the model results provide a suitable basis for evaluation of the impact of the scenario on expected annual dose.

3.10.2.2.1 Behavior of the Dike and Conduit

As described in *Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563]) an intrusion is assumed, for the purposes of the TSPA-SR, to be basaltic magma rising from a deep source in the form of one or more vertical tabular bodies of uncertain width, length, and azimuth. The intrusion is generally referred to as a dike, but it may actually be a set, or swarm, of multiple dikes that intrude essentially simultaneously. Each intrusive event (i.e., a swarm of one or more dikes) is assumed to generate one or more volcanoes somewhere along its length, but eruptions need not occur within the potential repository footprint. Based on the analysis documented in *Characterize Framework for Igneous Activity at Yucca Mountain (T0015)* (CRWMS M&O 2000 [141044]), 50 percent of dike intrusions will result in one or more eruption within the repository footprint (see Table 3.10.2). As described in *Igneous Consequence Modeling for TSPA-SR,* Section 6.1.2.9 (CRWMS M&O 2000 [139563]), adjusting this number to account for the area within the footprint that is occupied by waste-emplacement drifts yields approximately a 36 percent probability that an intrusive event that intersect the potential repository will result in one or more (up to a maximum of 5) eruptive conduits that intersect waste. The number of eruptive conduits is independent of the number of dikes in a swarm.

As shown in Figure 3.10-5, the eruptive conduit appears to form at the potential repository elevation. This representation is unrealistic, in that dikes are likely to rise relatively close to the surface before they focus into eruptive centers from which conduits propagate downward, but the figure is conceptually consistent with the simplifying assumption made in the TSPA that the effects of the dike and the conduit on the potential repository are independent. For the purposes of TSPA modeling of the eruptive scenario, all conduits within the repository footprint exist at the repository elevation, regardless of the elevation at which they form. For modeling of the intrusive scenario, all dikes that intersect the potential repository are also assumed to exist at the repository elevation, regardless of the elevation at which flow focuses into an eruptive conduit. Fragmentation of liquid magma into a pyroclastic flow is assumed to occur within eruptive conduit below the potential repository depth regardless of the depth at which the conduit forms. This assumption is consistent with analyses of water content of basalts in the Yucca Mountain region suggesting that volatile phases will exsolve at pressures corresponding to depths below that of the potential repository (CRWMS M&O 2000 [142657], Section 6.2.2).

Table 3.10-4. Process Model Factors and Associated Parameters for Volcanic Eruption

Factor	TSPA-SR Parameter	Distribution Type	Distribution or Value
Probability of Volcanic Eruption	Igneous Event Probability	CDF	Mean = 1.6×10^{-8} /yr.
	Probability of >0 Conduit in repository given an igneous event	Point Value	0.36

Table 3.10-4. Process Model Factors and Associated Parameters for Volcanic Eruption (Continued)

Factor	TSPA-SR Parameter	Distribution Type	Distribution or Value
Eruption Characteristics	Conduit Diameter	Log-normal	15 – 150 m, median = 50 m
	Event Eruptive Volume	Log-uniform	0.002 – 0.44 km^3
	Event Power	CDF	$1 \times 10^9 - 6.3 \times 10^{13}$ W
	Initial Eruption Velocity	CDF	2196 – 22294 cm/s
	Event Duration	CDF	33 minutes – 73 days
	Ash Mean Particle Diameter	Log-triangular	0.001 – 0.1 cm, mode = 0.01 cm
	Ash Particle Size Standard Deviation	Uniform	1 – 3 phi
Repository Response to Volcanic Eruption	Number of Packages Hit per Drift (Volcanic Eruption)	CDF	3 – 29, median = 10
	Number of Drifts Hit per Conduit	CDF	1 – 2, conduits with diameter ≥ 80 m may hit two drifts
	Number of Conduits Intersecting Waste	PDF	1 – 5, Probability of 1 = 0.86
	Percent of Hit Packages that Fail (Volcanic Eruption)	Point Value	100 %
	Waste Particle Size	Log-Triangular	0.0001 – 0.05 cm, mode = 0.002 cm
Atmospheric Transport in Volcanic Ash Plume	Particle Shape Factor	Point Value	0.5
	Constant (C) Relating Eddy Diffusivity and Particle Fall Time	Point Value	400 $cm^2/s^{5/2}$
	Incorporation Ratio	Point Value	0.3
	Ash Dispersion Controlling Constant	Log-uniform	0.01 – 0.5
	Ash Settled Density	Point Value	1.0 g/cm^3
	Wind Speed	CDF	~ 0 – 2000 cm/s, median = ~ 650 cm/s
	Wind Direction	PDF	Variable or fixed
Volcanic Biosphere	Soil Removal Factor	uniform	0.06 – 0.08 cm/yr
	Volcanic Eruption Biosphere Dose Conversion Factors	CDF	Variable, see Sec. 3.10.3

Source: CRWMS M&O 2000 [139563] (for all parameters except BDCFs and soil removal factor); CRWMS M&O 2000 [136281](for soil removal factor);
DTN: MO0006SPAPVE03.001 [151768] (for BDCFs) (The data in this DTN were preliminary at the time of the analyses and have since been updated. However, changes to the data are measurably minor and have no significant impact on any analyses reported in this document.)

NOTE: PDF = probability density function

Characteristics of the eruption such as eruptive power, style (violent versus normal), velocity, duration, column height, and the total volume of erupted material are included in the analysis through the ASHPLUME version 1.4LV code (CRWMS M&O 1999 [150744]). This code does not attempt to model the subsurface physics of the igneous event, but instead uses input parameters that characterize these properties to calculate the material available for atmospheric dispersal following a violent strombolian eruption. Earlier versions of the ASHPLUME code used by the Center for Nuclear Waste Regulatory Analyses for past analyses of Yucca Mountain (Jarzemba et al. 1997 [100987]) used inputs of event duration and event power to calculate event volume and column height. ASHPLUME version 1.4LV uses total erupted volume as the primary independent input parameter that characterizes the energy of the volcano, and calculates event duration and column height. This modification allows the code to sample from a distribution of event volumes that is based on observations of both past eruptions in the Yucca Mountain region (for which direct observations of duration and power are unavailable) and modern analog events. Observations of both modern and past analog volcanoes indicate that the violent phases account for only a portion of the total eruption, but available data do not support quantification of the ration of violent to nonviolent phases in potential future eruptions at Yucca Mountain. For the purposes of the TSPA-SR, the entire volume of erupted material in the analog volcanoes is assumed to have been ejected during a violent phase. This assumption is unrealistic, but ensures that the energy and size of future eruptions are not inappropriately underestimated in the TSPA-SR.

The number of waste packages damaged by a single eruptive conduit is determined by the diameter of the conduit and its location within the potential repository footprint (CRWMS M&O 2000 [139563], Section 6.1.2.9). Conduit diameter is a sampled parameter in the TSPA-SR calculation, with a distribution based on examination of analog sites that ranges from 15 to 150 m, with a median value of 50 m. Conduits that are less than 80 m in diameter (the approximate distance between drifts) can only intersect a single drift, and all waste packages that are fully or partially within the region intersected are assumed to be sufficiently damaged that they provide no further protection (Figure 3.10-8). Conduits that are greater than 80 m in diameter have an increasing probability of intersecting two drifts and damaging more packages. Multiple conduits associated with a single intrusion are assumed to have identical properties: i.e., a range of consequences are calculated for a single conduit, and if sampling selects more than one eruptive conduit within the repository footprint, consequences are scaled accordingly.

3.10.2.2.2 Behavior of the Waste Package and Waste Form in the Eruptive Conduit

For the purposes of the TSPA-SR, the contents of all packages that are fully or partially damaged by an eruption (i.e., that lie in part or entirely within the circumference of the conduit) are assumed to be fully available for entrainment in the eruption. Waste packages, drip shields, cladding, and all other components of the EBS are assumed to be sufficiently damaged that they provide no further protection for the waste.

This assumption provides a conservative bound to the possible behavior of the engineered barriers in the eruptive conduit in the absence of defensible models and data to support a more realistic treatment of the processes. The degree to which this assumption causes an overestimation of the eruptive releases has not been quantified, but available information indicates that packages are likely to fail. Actual conditions in the conduit and the response of the engineered barriers are uncertain, but temperature alone may be sufficient to cause failure of the waste packages. Magmatic temperatures for Yucca Mountain region basalts are estimated to be approximately 1,050 to 1,170°C (CRWMS M&O 2000 [142657], Section 6.2.4). Calculations of the waste package strength at high temperatures indicate that failure of end cap welds will occur at 1,200°C from thermal stresses alone (CRWMS M&O 1999 [121300]). Mechanical and chemical conditions in the conduit will further contribute to degradation of the waste packages.

Physical properties of the waste in the eruptive environment are estimated based on the assumption that the waste form is directly exposed to the eruption, consistent with the assumption that the waste packages provide no protection. As discussed in the following section, the extent to which waste particles are entrained in the eruption is controlled by an incorporation ration based on the ratio of the waste particle diameter to the ash particle diameter, with smaller waste particle sizes resulting in greater entrainment and transport. The range of waste particle diameters used in the TSPA-SR, 1 to 500 microns with a mode of 20 microns, is based on laboratory observations of particle sizes of unaltered spent nuclear fuel following mechanical grinding (CRWMS M&O 2000 [146498]). All waste in damaged packages is assumed to exist as particles in this size range throughout the eruption, neglecting the time required to degrade the waste package and fuel rods. (The assumption of instantaneous degradation is conservative in principal, but is a reasonable simplification that will have little or no effect on eruptive releases if the eruption continues long enough for full degradation to occur.) Other waste forms may have different particle diameters in the eruptive environment, depending both on the initial type of the waste (commercial spent fuel or glass waste) and the degree and type of alteration of the waste. Treating all waste as unaltered commercial spent fuel is conservative with respect to the unaltered glass waste forms because this waste is likely to have particle diameters comparable to those of the ash, larger than the values used for spent fuel. The assumption that the waste form is unaltered is reasonable for analyses of the 10,000-year postclosure performance period, given the relatively small number of waste packages expected to fail under nominal conditions during that period and the expected stability of the waste form within the undisturbed waste packages (CRWMS M&O 2000 [139563], Section 5.3.5).

3.10.2.2.3 Modeling Transport of Waste in the Ash Plume

The volcanic eruption and subsequent transport of radioactive material in the ash plume are modeled using ASHPLUME version 1.4 LV (CRWMS M&O 1999 [150744]), which is a modification of ASHPLUME version 1.0, developed at the Center for Nuclear Waste Regulatory Analyses (Jarzemba et al. 1997 [100987]). The model is an implementation of an approach developed by Suzuki (1983 [100489]) that uses a parametric characterization of the properties of the eruption to calculate a source term of ash for atmospheric transport. For a single eruptive event, the code calculates particle dispersal and vertical settling from a vertical line source in a constant wind stream, yielding a concentration of ash particles at selected points on the ground downwind of the source (Figure 3.10-9). The quantity of ash deposited at any specified point is a function of the wind speed and direction, the volume of ash erupted (which determines the duration and height of the eruption), and the ash particle diameter. The ash particle diameter distribution used in the TSPA-SR, 0.01 to 1 mm, is based on observations from violent eruptions at modern analog volcanoes (CRWMS M&O 2000 [142657], Section 6.5.1).

No data are available regarding wind conditions at Yucca Mountain during the distant future, and the values for speed and direction are therefore sampled from distributions developed from past observations in the Yucca Mountain region (Quiring 1968 [119317], pp. VI-1 to VI-21). Speed and direction data from an eight-year period (1957 to 1964) are reported from 1,500 to 5,000 m (5,000 to 16,000 ft) above sea level for four different months of the year. As is typical of weather data, short-term variability is relatively high in this data, providing coverage of a broad range of uncertainty appropriate for the analysis of relatively brief volcanic events.

Although speed and direction are reported as paired data for each observation at each elevation, the two parameters are sampled independently in the TSPA-SR, allowing for a full coverage of wind speeds at each direction. Data from all elevations, time of year, and time of day are weighted equally in constructing the distributions, consistent with the ASHPLUME code's use of a single velocity at all elevations. This aggregation of data into a single distribution ensures that extreme values from higher and lower altitudes are fully represented in the analysis. Speeds range from 0 to approximately 2000 cm/s, with a median speed of approximately 650 cm/s. Direction is variable, consistent with the observed data, but the decision to treat direction and speed as uncorrelated parameters allows flexibility in the TSPA to consider cases in which the wind direction is fixed toward a specific location in all realizations, rather than allowed to vary (CRWMS M&O 2000 [139563], Section 5.1.2).

Waste particles are entrained in the eruption based on the ratio between the diameter of the ash particles and their diameter. For the incorporation ratio used in the TSPA-SR, waste particles one half the diameter of the ash, or smaller, are incorporated into the ash particles and transported downwind. The code then calculates downwind transport of waste-contaminated ash taking into account wind speed and gravitational settling (CRWMS M&O 2000 [139563]).

3.10.2.3 Total System Performance Assessment-Site Recommendation Model for Groundwater Transport of Radionuclides Following Igneous Intrusion

The TSPA-SR model for radionuclide mobilization and transport away from packages that have been damaged by igneous intrusion is described in *Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563]) essentially the same as the nominal model for radionuclide mobilization and transport, modified to include an igneous intrusion that intersects the potential repository (Figure 3.10-10). As shown in Figure 3.10-11, the igneous intrusion groundwater transport model uses information about the probability of intrusion, the characteristics of the intrusion, and the response of the potential repository to calculate damage to waste packages. Groundwater transport away from the damaged packages is calculated using the nominal scenario models, and doses to humans from contaminated groundwater are determined using nominal BDCFs. Igneous intrusion groundwater doses are weighted by the probability of intrusion to yield the probability-weighted dose needed to calculate the expected annual dose specified by proposed 10 CFR 63.113(b) (64 FR 8640 [101680]).

There is no separate set of computational models used in the TSPA to simulate the consequences of intrusion. Instead, the igneous intrusion groundwater transport model consists of a set of process model factors and associated input parameters (Table 3.10-5) used to define a modified source term for calculations using the flow and transport models developed for the nominal case. There are three main components to the model, (1) assumptions regarding the behavior of waste packages and other elements of the EBS that have been damaged as a result of proximity to an igneous intrusion, (2) assumptions regarding the use of the nominal models for groundwater flow and radionuclide transport away from the waste packages, and (3) calculation of the number of packages that are damaged. Figure 3.10-12 shows the key elements of the igneous groundwater transport scenario, major model inputs and outputs, and the major bases for confidence that the model results provide a suitable basis for evaluation of the importance of the scenario.

Table 3.10-5. Process Model Factors and Associated Parameters for the Igneous Intrusion Groundwater Transport Scenario

Factor	TSPA-SR Parameter	Distribution Type	Distribution or Value
Probability of Igneous Intrusion	Igneous Event Probability	CDF	Mean = 1.6×10^{-8} /yr.
Intrusion Characteristics	Number of dikes per event (number of dikes in a swarm)	Log-normal	Minimum = 1, mean = 3 95th percentile =10
	Dike width	Log-normal	Minimum = 0.5 m, mean = 1.5 m, 95th percentile = 4.5 m
Repository Response to Igneous Intrusion	Number of packages damaged on either side of a dike in each intersected drift (with backfill)	Point value	3
	Number of packages damaged on either side of a dike in each intersected drift (without backfill)	Point value	all
	Percent of damaged packages that Fail (Igneous Intrusion)	Point Value	100
	Number of Packages damaged (Igneous Intrusion with backfill)	CDF	105 – 227, median = 192
	Number of Packages damaged (Igneous Intrusion without backfill)	CDF	0 – 11180, median = 1720
Groundwater transport following Igneous Intrusion	Nominal performance models and parameters		
Biosphere	Nominal performance model and parameters		

Source: CRWMS M&O 2000 [139563]; CRWMS M&O 2000 [142657]

3.10.2.3.1 The Number of Waste Packages Damaged by Igneous Intrusion

Magma entering the drift will undergo rapid depressurization as the confining pressure drops from lithostatic to atmospheric. For most of the range of water contents estimated for Yucca Mountain region basaltic magmas (CRWMS M&O 2000 [142657]) depressurization may be accompanied by rapid exsolution of volatile phases and explosive fragmentation of the magma into pyroclasts. As discussed in *Dike Propagation Near Drifts* (CRWMS M&O 2000 [142635]), damage to the packages immediately adjacent to the point of intrusion is likely to be extensive. The force of the shock wave accompanying the fragmentation will be sufficient to move packages off their emplacement pallets, and to cause displacement of 3 or 4 packages on either side of the dike. Rather than use a variable number of packages that are partially damaged (3 or 4), the TSPA-SR input is based on a calculation in which 3 packages on either side of an intrusive dike are fully damaged in each drift that is intersected (CRWMS M&O 2000 [142663]). The number of drifts intersected is calculated probabilistically, based on the drift spacing and distributions for the azimuth and length of intrusive dikes. In the absence of data characterizing the spacing of future dikes at Yucca Mountain, multiple dikes in a swarm are assumed to be sufficiently far apart that they behave independently, with 6 packages damaged between them.

For potential repository design alternatives that include backfill, damage within a drift is limited to the 3 packages on either side of the intrusion. Backfill pushed up by displaced waste packages and debris from damaged drip shields will stop magmatic material as it moves away from the dike, and the drift is assumed to be plugged by debris and magma relatively close to the intrusion. Pressure will increase in the isolated section of the drift on either side of the dike to the lithostatic pressure of the magma (further damaging the waste packages), and the dike will continue to propagate upward. Damage will have been severe within the affected region, but will be relatively minor further down the drift, behind the plug of backfill and debris. For analyses of design alternatives that include backfill, the TSPA-SR assumes that packages beyond plug of backfill and debris are undamaged.

For the SR reference potential repository design, which does not include backfill, damage within the drift will be more extensive. Actual conditions are uncertain, but the shock wave following decompression of the magma could propagate the full length of the affected drift. Immediate mechanical damage from displacement of waste packages may be limited to the region adjacent to the point of intrusion, as in the backfill model, but extensive damage to the drip shield and ground support and some damage to the waste packages may occur throughout the drift. More importantly, debris from remains of the EBS will likely not be sufficient to create a plug anywhere before the right angle intersections at the ends of the drifts. Pyroclastic material (or liquid lava, in the possible case of an extremely dry magma) will quickly fill the entire length of the drift, and pressure will rise from atmospheric to lithostatic before the dike can continue to propagate upward. The combination of high temperature (approximately 1,040 to 1,170°C) and high pressure (approaching the magmatic lithostatic pressure of 7.5 MPa at the potential repository depth) will be more than sufficient to cause failure of the packages (CRWMS M&O 2000 [142635]). Therefore, for the no-backfill design, the TSPA-SR assumes that all packages in drifts that are intersected by intrusive dikes are damaged by the intrusive event. The number of drifts intersected is calculated probabilistically, based on the drift spacing and distributions for the azimuth and length of intrusive events. Based on an average number of 215 waste packages per drift and a median number of 8 drifts intersected per intrusion, the median number of packages damaged per intrusion is 1,720. Approximately 10 percent of realizations result in damage to 5,000 or more packages. As discussed in the following section, 3 packages on either side of the dike are assumed to be sufficiently damaged that they provide no further protection for the waste, as in the backfill case, and the remaining packages in each intersected drift are assumed to undergo end cap weld failure.

3.10.2.3.2 Waste Package Behavior in the Intrusive Environment

Waste package behavior in immediate vicinity of the intrusion is bounded by the assumption that 3 packages on either side of dike are sufficiently damaged that they provide no further protection for the waste. As is the case for the eruptive environment, actual conditions are uncertain, and damage is likely to range from moderate to extensive. Complete destruction of these waste packages seems unlikely, but thermal stresses alone may be sufficient to cause failure of the end caps (CRWMS M&O 1999 [121300]), and there is insufficient evidence to support a less conservative approach to the package behavior given the likely mechanical stresses and elevated pressures. Drip shields and cladding are also assumed to provide no further protection for the waste in the region adjacent to the dike.

If backfill is present, damage is assumed to be limited to the region containing the 3 packages on either side of the dike. For the SR reference potential repository design without backfill, all remaining waste packages in all drifts intersected by a dike are assumed to be breached with a hole of uncertain cross-sectional area, and all drip shields and cladding in the intersected dikes are assumed to be fully destroyed. Breaching of the waste packages is consistent with the analysis reported in *Dike Propagation Near Drifts* (CRWMS M&O 2000 [142635]) which concludes that end cap welds will fail on these packages due to high temperatures and pressures. The area of the hole created by end cap weld failure represents the cross-sectional area that might open in a failed weld before gas flow into the failed package equalizes internal and external pressures, halting the propagation of the crack. This value is uncertain, and sampled from a log-normal distribution with a mean value of 10 cm^2 (CRWMS M&O 2000 [139563]). The minimum value of the distribution is 1 cm^2, and the maximum is 1.9×10^4 cm^2, which is an approximation of the full cross-sectional area of a representative end cap with a radius of 77 cm. Although the mean value can be thought of conceptually as corresponding to a 1-mm-wide crack that propagates for 1 m along a weld, or a 2 mm-wide crack that extends 50 cm, it was not chosen to represent any specific dimensions of a weld failure because failures are likely to have variable shapes. Rather, it was chosen as a conservative approximation of the size of opening necessary to permit rapid gas flow and pressure equilibration. Sampling the area of the breach from a distribution that includes much larger hole sizes is intended to account for both uncertainty regarding the nature of the magmatic fluids and the package response and spatial variability in the extent of damage within the drifts.

3.10.2.3.3 Radionuclide Transport following Igneous Intrusion

As described in *Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563], Section 3.4), nominal models for radionuclide mobilization and transport are used in lieu of a separate set of detailed source term, flow, and transport models for the conditions in the drift following intrusion. All waste in damaged packages is available for transport in the UZ, dependent on solubility limits and the availability of water, which is determined using the seepage model for nominal performance. Thermal, chemical, and mechanical effects of the intrusion on the drift environment are neglected for transport modeling. No credit is taken for water diversion by the remnants of the drip shield, and cladding is assumed to be fully degraded. Remnants of waste packages within the zone of extensive damage adjacent to the dikes are assumed to provide no barrier to flow or radionuclide transport. Actual thermal, chemical, hydrological, and mechanical conditions in the drift following igneous intrusion are unknown, but the conservatism of assuming complete failure of engineered barriers in the zone of greatest damage and partial failure elsewhere in intruded drifts is considered sufficient to compensate for uncertainty associated with conditions in the drift.

3.10.3 Exposure Pathways and the Biosphere Dose Conversion Factors for the Igneous Disruption Scenarios

The TSPA-SR biosphere model described in Section 3.9 has been adopted without change for the igneous intrusion groundwater release pathway, because this release mechanism introduces no new exposure pathways beyond those considered in the nominal scenario. BDCFs for igneous groundwater releases are identical to those described in Section 3.9 for the nominal scenario.

The biosphere model requires considerable modification for conditions following a volcanic eruption, however, because it must consider radionuclide transport and uptake following the deposition of contaminated volcanic ash. As shown schematically in Figure 3.10-13, biosphere processes that must be considered after the ash is deposited include ash resuspension, redistribution, and erosion, as well as radionuclide uptake in plants and animals. Human exposure may occur as a result of inhalation of fine particles of contaminated ash during the eruptive event, inhalation of resuspended ash after deposition, ingestion of larger particulates after inhalation both during and after the event, and ingestion of contaminated crops and animal products. Consumption of contaminated water, which is an important pathway for the nominal scenario and the igneous intrusion groundwater transport scenario, is not included a pathway for volcanic eruption because there is no significant surface water in the Yucca Mountain region that might be contaminated by volcanic ash.

3.10.3.1 Exposure Pathways during Volcanic Eruption

Ash concentrations in the air may be extremely high during the eruptive event, and humans who do not leave the region may be exposed to radiation by both inhalation and ingestion of particulates. Dose factors that account for both inhalation and ingestion pathways have been developed for the relatively brief period of the eruptive event (between 33 minutes and 73 days, from *Characterize Eruptive Processes at Yucca Mountain, Nevada* (CRWMS 2000 [142657], Table 5). Dose factors for each radionuclide represent doses per day resulting from exposure to air containing unit activity concentration of the specific radionuclide. One third of the radioactivity in the air is assumed to consist of the respirable fraction of particles smaller than 10 microns (PM_{10}), and two-thirds of the radioactivity is assumed to be contained in larger particles up to 100 microns that are swallowed following inhalation, thus contributing to an ingestion dose. Dose factors are based on an assumption of a breathing rate of 23 m^3/day of air (DTN: MO0006SPAPVE03.001 [151768]).

Dose factors for exposure during the eruption phase are summarized in Table 3.10-6. The use of dose factors to calculate dose is as follows:

$$Dose\ Rate = PM_{10}\left(\frac{g}{m^3}\right) \times C_{ash}\left(\frac{pCi}{g}\right) \times Dose\ Factor \qquad \text{(Eq. 3.10-1)}$$

where: PM_{10} = mass concentration of PM-10 fraction of suspended particulates, (g/m^3)
C_{ash} = activity concentration of radionuclide in ash, (pCi/g).

Table 3.10-6. Dose Factors for Exposure During the Eruption Phase

Dose Factors, rem/d per pCi/m^3	
^{227}Ac	6.92E-02
^{241}Am	4.64E-03
^{243}Am	4.60E-03
^{137}Ce	1.36E-06
^{231}Pa	1.34E-02
^{210}Pb	2.07E-04
^{238}Pu	4.10E-03
^{239}Pu	4.49E-03
^{240}Pu	4.49E-03
^{242}Pu	4.29E-03
^{226}Ra	1.16E-04
^{90}Sr	5.39E-06
^{229}Th	2.22E-02
^{230}Th	3.36E-03
^{232}U	6.80E-03
^{233}U	1.40E-03
^{234}U	1.37E-03

Source: DTN: MO0006SPAPVE03.001 [151768]

Mass concentrations of ash suspended in air following an eruption at Yucca Mountain are uncertain, and may be quite high. Data are not available for air mass loading during basaltic eruptions like those that have occurred in the Yucca Mountain region, but concentrations reported for other types of eruptions are high. For example, data from the May 1980 eruption of Mount St. Helens in Washington State (a massive silicic eruption that is not a good analog for Yucca Mountain basaltic volcanism) indicates that total suspended particulate concentrations (including both PM_{10} and larger particles) in Yakima, approximately 135 km from the volcano, ranged from 5,800 to 13,000 micrograms/m^3 during the first five days after the eruption (CRWMS M&O 2000 [149736], p. 36). However, these measurements represent conditions during the relatively short period of actual ashfall (i.e., days) and ensuing high levels of resuspension due to traffic, ash removal efforts, and wind. Humans are unlikely to tolerate exposures to airborne concentrations this high for more than brief periods of time before seeking shelter. Realistic air mass loading values applicable to the average member of the critical group, even during the eruptive event, are considerably less because they should represent exposure averaged over a full day, rather than instantaneous values. The TSPA analysis assumes that the concentration of respirable particulates (PM_{10}) that the average member of the critical group will be exposed to for the full duration of the eruptive event is 1000 micrograms/m^3 (DTN: MO0006SPAPVE03.001 [151768]. The data in this DTN were preliminary at the time of the analyses and have since been updated. However, changes to the data are measurably minor and have no significant impact on any analyses reported in this document.). This value is significantly higher than the EPA's PM_{10} breakpoint of 604 micrograms/m^3, for the highest Air Quality Index of 500 (40 CFR Part 58 [150242], Appendix G, Table 2). An Air Quality Index value of 500 corresponds to the significant harm level, at which serious and widespread health effects occur to the general population, and is considered a reasonable value for the highest concentration an average member of the critical group would be exposed to during volcanic eruption.

As described in Section 5.2.9, doses calculated using these factors are significantly below the eruptive dose from other pathways, and do not significantly contribute to the calculated expected annual dose. Therefore, the eruptive phase doses are not included in the TSPA-SR results described in Section 4.2.

3.10.3.2 Ash Redistribution, Resuspension, and Soil Processes

Immediately following deposition, the ash layer will be unconsolidated material, and will be subject to resuspension and redistribution by wind and water action. As time passes, the ash layer will stabilize, and in the absence of further disruption, it will eventually become a relatively stable soil that is less susceptible to resuspension. Both the amount of redistribution of the ash and the time required for stabilization of the ash layer are uncertain, and neither process has been addressed explicitly in the TSPA-SR.

However, effects of surface redistribution of the ash layer have been approximated in the TSPA by fixing the wind direction at the time of eruption to the south, toward the critical group. This assumption is not intended to directly simulate surface redistribution, but provides a conservative approach that essentially maximizes the effects of wind-borne redistribution by placing the critical group on the centerline of the plume, where ash thickness is greatest. Airborne redistribution is as likely to remove ash from the location of the critical group as it is to deposit it, and there is no credible atmospheric mechanism for preferentially generating deposits along the centerline of the plume thicker than those that were initially deposited. Surface redistribution will also occur by water action and could result in thicker deposits in regions where sediments that are transported down Forty-Mile Wash are deposited. This possibility is accounted for in biosphere modeling by the development of alternative BDCFs for ash layers 15 cm and above in thickness, as discussed in the following section. Although most ash layers are expected to be relatively thin at the location of the critical group (1 cm or less), thicker deposits are possible, either from direct atmospheric deposition or water transport, and have been considered in the volcanic biosphere model (DTN: MO0006SPAPVE03.001 [151768]).

Once deposited, ash layers are subjected to normal soil processes, including erosion that will tend to remove radioactive material from the soil surface. For stabilized soil, erosion processes may be relatively slow. For plowed agricultural soil, however, soil removal processes are relatively rapid. Because characteristics of agricultural soil support a greater level of human exposure than unplowed land (due both to contamination of crops and inhalation of ash resuspended from the plowed land), the TSPA-SR uses a soil removal factor that is consistent with cropland. Soil is assumed to be eroded at a rate of 0.06 to 0.08 cm/yr, based on site-specific data published by the United States Department of Agriculture (CRWMS M&O 2000 [136281]). As discussed in the following section, BDCFs are used that are based on high air mass-loading values, consistent with resuspension of soil from plowed land.

3.10.3.3 Exposure Pathways from Contaminated Ash

In addition to exposure during the eruptive event, biosphere modeling considers exposure both during a transition period when ash is easily mobilized and during a later steady-state period in which ash is stabilized and contaminants are mixed into soil. Exposure is considered both from relatively thin ash layers (1 cm or less in thickness) and from thicker deposits possible from larger eruptions (15 cm and greater in thickness). Radionuclides are assumed to be uniformly distributed within the ash layers, regardless of their thickness. During the transition period, radionuclides are assumed to remain within the ash layer, rather than mixing into the underlying soil. This assumption has no effect on radionuclides concentrations in the thick ash layers, but maximizes concentrations in the near-surface soil for thin ash layers, increasing the potential for resuspension and human exposure by inhalation during the transition period. For the steady-state period, ash and radionuclides are assumed to be fully mixed within a 15-cm soil layer.

As discussed in the previous section, soil erosion is assumed to occur on agricultural land, removing a thin layer from the upper surface of the soil each year. For both the transition and steady-state periods, the radionuclides that remain are assumed for the purposes of calculating soil loss due to erosion to be mixed into a 15-cm layer. For the transition period, this assumption is inconsistent with the assumption of no mixing used for calculation of the resuspension pathway; the inconsistency is conservative, because it prevents rapid removal of thin ash layers while keeping surface concentrations high.

Conditions for human exposure are the same as those considered for nominal performance. That is, the human receptor is represented by an average member of the critical group and is assumed to live year-round in the farming community. The receptor is also assumed to be involved in the activities typical of the current inhabitants of the region (e.g., has the same patterns of consumption of locally produced foods and time spent in outdoor activities).

3.10.3.4 Input Parameters for the Volcanic Eruption Scenario Biosphere Modeling

Input parameter values to the GENII-S computer program (CRWMS M&O 1998 [107723]) used to calculate BDCFs for the volcanic eruption scenario are similar to those used for the nominal-scenario biosphere modeling (Section 3.9.3), with the exception of the parameters used to characterize airborne concentrations of radionuclides. These parameters are adjusted for the eruption scenario to account for the increased dustiness following an eruption. Parameters related to dustiness are the total suspended particulate concentration and the respirable fraction (also called the mass load). Total suspended particulates are the entire amount of particulate matter in the air; this parameter is not used directly in the biosphere modeling, but is used to calculate the crop resuspension factor (a measure of the amount of particulate matter that is resuspended and available for deposition on plant's surfaces). Mass load is the respirable fraction of the total suspended particle that is small enough (i.e., PM_{10}, less than 10 microns) to be inhaled into the lungs. Related to these parameters is the parameter for inadvertent soil ingestion, which includes ingestion of particles that can be trapped in the nasal passages and airways and subsequently passed to the ingestion pathway. Therefore, this parameter could increase with increased dustiness.

Parameter distributions for total suspended particulates and air mass loading following a volcanic eruption are consistent with data collected following the Mount St. Helens eruption (which was a massive silicic eruption that involved far greater quantities of ash than are associated with the Yucca Mountain region basaltic volcanoes) and are defined as follows. As discussed in Section 3.10.2.1, during the eruption, the mass load is assumed to 1,000 micrograms/m^3. During the transition period, air mass load is assumed to be uncertain, and ranges from 1000 micrograms/m^3 to 30 micrograms/m^3, which is the average annual value for the State of Nevada (DTN: MO0006SPAPVE03.001 [151768]). As the upper value was unsustainable around Mount St. Helens, a log-uniform distribution, emphasizing the lower values between these bounds, is used. Total suspended particle is assumed to be 3 times higher than the mass load. The crop resuspension factor based on the total suspended particle thus has a similar distribution to the mass load. Soil ingestion, a constant in the nominal-scenario modeling, can be estimated from average breathing rate and time outdoors and could increase because of the increased total suspended particle by 20 mg/day. In the modeling, it is log-uniformly distributed between 50 and 70 mg/day. Table 3.10-7 presents these parameters and their distributions.

Table 3.10-7. Input Parameters Used in Biosphere Modeling Specifically for the Volcanic Eruption Scenario

Parameter	Distribution
Inhalation Exposure Mass Load (g/m^3)	Log uniform (3×10^{-5},1×10^{-3})
Soil Ingestion Rate (mg/day)	Log uniform (50, 70)
Crop Resuspension Factor (/m)	Log uniform (9×10^{-9}, 3×10^{-7})

Source: DTN: MO0006SPAPVE03.001 [151768]

3.10.3.5 Biosphere Dose Conversion Factors for Volcanic Eruptive Releases

The biosphere modeling for the volcanic direct-release mechanism involves construction of BDCFs (Section 3.9). As with the case of BDCFs for the nominal, undisturbed scenario, calculations of these conversion factors for the volcanic direct-release case used GENII-S (CRWMS M&O 1998 [107723]) in a series of probabilistic runs to propagate the uncertainties of input parameters into the output. A Latin Hypercube sampling technique (a form of stratified sampling) is used in a statistical analysis of 160 realizations to generate 160 estimates of annual dose caused by unit areal concentration of each relevant radionuclide. These dose estimates are used to create BDCFs in the form of discrete cumulative probability distributions that are entered in the TSPA model. BDCFs have been calculated for 1 cm and 15 cm ash layers for both the transition and steady-state periods following a volcanic eruption (DTN: MO0006SPAPVE03.001 [151768]). The TSPA-SR uses the transition BDCFs to calculate doses for all time following a volcanic eruption, neglecting soil stabilization and maintaining relatively high air mass loading indefinitely. The TSPA-SR also uses BDCFs calculated for a thin (1-cm) ash layer rather than for the thick layer, consistent with ASHPLUME (CRWMS M&O 2000 [151349]) results (see Section 3.10.4) indicating that median ash layer will be relatively thin. This assumption is potentially conservative with respect to the treatment of thicker ash layers, because all radionuclides in thicker layers are assumed to be concentrated in the upper 1 cm, rather than being mixed throughout the full layer thickness. The impact of these assumptions is discussed further in Section 5.2.9.

The mean and standard deviations of the distributions for the thin-layer transition BDCFs are given in Table 3.10-8. These are the BDCFs used in the TSPA-SR to generate the dose histories shown in Section 4.2. Alternative BDCFs for thicker layers and for steady-state soil conditions are described in DTN: MO0006SPAPVE03.001 [151768].

Table 3.10-8. Biosphere Dose Conversion Factors for Volcanic Eruptive Release of Radionuclides

Radionuclide	BDCF (rem/yr. per pCi/ m^2)	
	Arithmetic Mean	Arithmetic SD
^{90}Sr	1.22E-8	1.91E-8
^{137}Ce	1.28E-9	1.52E-9
^{210}Pb	6.05E-8	6.68E-8
^{226}Ra	5.66E-9	3.42E-9
^{227}Ac	7.34E-7	6.46E-7
^{229}Th	2.31E-7	2.06E-7
^{230}Th	3.47E-8	3.09E-8
^{231}Pa	1.63E-7	1.24E-7
^{232}U	7.39E-8	6.45E-8
^{233}U	1.50E-8	1.30E-8
^{234}U	1.48E-8	1.28E-8
^{238}Pu	4.94E-8	3.78E-8
^{239}Pu	5.48E-8	4.19E-8
^{240}Pu	5.47E-8	4.19E-8
^{242}Pu	5.11E-8	3.91E-8
^{241}Am	5.60E-8	4.27E-8
^{243}Am	5.59E-8	4.26E-8

Source: DTN: MO0006SPAPVE03.001 [151768]

As shown in the table, 17 radionuclides have been identified as relevant for calculation of BDCFs under the volcanic direct-release case (CRWMS M&O 2000 [147096], Section 7.1). The list differs from that considered for the nominal scenario because it reflects the radionuclide inventory directly released from the potential repository during a volcanic eruption, as opposed to radionuclides transported to the biosphere by groundwater in the SZ where substantial retardation and decay occurs within the geologic strata.

3.10.3.6 Integration into the Total System Performance Assessment Model

BDCFs are expressed in units of rem/yr. per pCi/m^2 (the units of the BDCFs for the nominal, undisturbed performance scenario are rem/yr. per pCi/L; Section 3.9). Within the Total System Performance Assessment model, the areal mass of a radionuclide in the ash deposited on the ground surface is calculated, considering radioactive decay and soil removal. The areal mass is then converted to an areal activity; and the areal activity is multiplied by the appropriate BDCF to realize the annual TEDE (in units of rem/yr; here called the annual dose) for that radionuclide.

Within the TSPA model, at every time step, the annual doses for all the radionuclides are summed to determine the total annual dose. The total annual doses from all of the realizations are averaged to determine the mean annual dose for each time step. Doses are calculated separately for the volcanic eruptive release mechanism, igneous intrusion groundwater release mechanism, and the nominal scenario. These doses are combined as described in Section 4.3 to achieve the expected annual dose to the receptor.

3.10.4 Treatment of Uncertainty and Variability in the TSPA Model for Igneous Disruption

Uncertainty and variability regarding the probability and consequences of future igneous events at Yucca Mountain have been incorporated into the TSPA through several approaches, as discussed throughout Section 3.10. In many cases, uncertainty and variability have been included through the use of parameter distributions that allow a range of values to be used in the simulations. These distributions are described in Tables 3.10-4 and 3.10-5. In other cases, where data are insufficient to support realistic models or a defensible distribution of parameter values about a best-estimate value, the TSPA-SR relies on conservative assumptions that ensure that the analysis does not underestimate the impact of the phenomenon. As summarized here, this treatment of uncertainty and variability provides confidence that the TSPA-SR analysis provides a reasonable and appropriate basis for evaluating the potential consequences of igneous disruption at Yucca Mountain.

Uncertainty and variability in the probability of igneous disruption–As described in Section 3.10.1, the probability of future igneous disruption at Yucca Mountain is based on expert elicitation documented in the Probabilistic Volcanic Hazard Analysis (CRWMS M&O 1996 [100116]). The experts involved in this elicitation explicitly considered uncertainty in current understanding of volcanic processes and regional geology in making their estimates. The analysis took into account both spatial and temporal variability, and the resulting distribution of event probabilities reflects the full range of uncertainty expressed by the experts. The mean annual event probability used in the TSPA-SR (1.6×10^{-8}) represents the expected value of the range of values estimated by the experts, adjusted for the current repository design layout (CRWMS M&O 2000 [141044]). The experts considered the possibility that the rate of volcanic activity in the future might increase or decrease, and this temporal variability was included in their range of estimates of the annual frequency of igneous events. The range of values used in the TSPA-SR therefore accounts for temporal variability, although the annual probability used in any single realization is held constant after it is selected by sampling.

Uncertainty and variability in the modeling of the consequences of igneous disruption–As described in Section 3.10.2, the TSPA-SR relies on both uncertainty distributions for selected input parameters and conservative modeling assumptions to ensure that uncertainty and variability regarding the effects of igneous disruption have been adequately included. Examples of uncertainties that have been included through parameter distributions include eruption size and power (by using a range of values for the volume of erupted volume), wind speed, ash and waste particle sizes, the number of packages damaged in an igneous event, and other parameters listed in Tables 3.10.4 and 3.10.5. Spatial and temporal variability regarding these parameters is, in general, not described explicitly in these parameter distributions, but is instead included in the analysis through the use of multiple realizations using fixed values. For example, the data set used to develop the distribution for wind speed included data regarding both speed and direction from different times (including seasonal variability) and different altitudes. The TSPA-SR model used this data to develop separate distributions for speed and direction that used all values independently of altitude and time, creating a distribution of wind speeds that includes the full range, from slowest to highest values (CRWMS M&O 2000 [139563], Section 5.1.2). Using values from this range as constants for all altitudes and times creates a distribution of conditions with a mean that is a reasonable approximation of the mean of the true conditions, but which provides better resolution of the importance of extreme values. For example, this treatment of variability allows the possibility for high-speed winds characteristic of high altitudes to occur at all altitudes throughout the eruptive event.

Conservative assumptions have been used in several places in the consequence analysis where data are insufficient to support implementation of more realistic models. In general, it is not possible to quantify the impacts of these conservatism: if data were available to quantify the impact, a more realistic assumption would have been used. Examples of bounding assumptions of this type occur in the treatment of the response of the waste package to igneous disruption. The assumption that all waste packages in the direct path of an eruptive conduit are sufficiently damaged that they provide no further protection for the waste is an example of an assumption that is surely bounding (damage to these packages can be no greater than this), but data are not available to support a more realistic model. Were such data hypothetically available, they might show that uncertainty regarding actual conditions should include a range of waste package responses, perhaps including some continued protection for the waste form. In the absence of such data, the TSPA-SR simply adopts the bounding endpoint of the unknown distribution. Similarly, the assumption that the three waste packages on either side of an intrusive dike are sufficiently damaged that they provide no further protection to the waste provides a bound to the performance of those packages. In this case, damage is unlikely to be as severe as that suffered by the packages directly in the path of an eruption, and a more realistic distribution would likely include some continued protection of the waste form. The TSPA-SR assumption provides a sure bound for the performance of these packages, and through its conservatism provides additional compensation for uncertainty regarding the extent of damage to packages further from the point of intrusion.

Uncertainty and variability in the treatment of biosphere pathways following igneous eruption–As described in Section 3.10.3, sets of BDCFs have been developed for the TSPA-SR appropriate for a range of possible conditions following an eruption. These sets of BDCFs provide a reasonable estimate of uncertainty regarding the appropriate values to use in the TSPA. Specifically, BDCFs are available for both relatively thin (1 cm or less) ash layers and relatively thick layers (15 cm or greater), for conditions that might occur in the first decade following eruption (the "transition phase" in which ash is relatively easily remobilized) and also for long-term steady-state conditions. Rather than attempting to include the full range of uncertainty included in these multiple sets of BDCFs, the TSPA-SR uses a single set of BDCFs (the thin-layer transition phase values) that is conservative with respect to the alternatives. This set of BDCFs places all radionuclides in the ash layer in the upper 1 cm, regardless of the calculated thickness of the layer, and assumes that air mass loading remains high permanently above the ash deposit, overestimating the long-term effects of the inhalation pathway.

The treatment of uncertainty and variability in the biosphere contains an important example of the use of conservative assumptions even when more realistic data are available. Independent of other factors, the decision to use the transition-phase BDCFs for all times following eruption could be interpreted to be unnecessarily conservative, because more realistic BDCFs are available to describe the steady-state conditions associated with stabilized ash and soil layers. Similarly, the assumption to neglect spatial variability in wind direction by fixing the wind direction toward the critical group could be interpreted as unnecessarily conservative because data are available to support a realistic characterization of wind directions. Both of these conservative assumptions are included in the TSPA-SR to provide a reasonably conservative approach to compensating for uncertainty regarding surficial processes that might move contaminated ash from its initial point of deposition to the location of the receptor. Surficial transport processes, which are not modeled explicitly in the TSPA-SR, could potentially result in contaminated ash reaching the location of the receptor by resuspension in wind or following intermittent flooding in Forty-Mile Wash. Sediment transport in Forty-Mile Wash is of particular concern, because much of the total volume of contaminated ash that could be erupted by a volcanic conduit that intersects the potential repository is likely to fall somewhere within the Forty-Mile Wash drainage basin, regardless of the wind direction during the event. Some fraction of this sediment will eventually be carried down the wash by water, and could be redeposited in the vicinity of the receptor group. The assumption in the TSPA-SR that the wind direction is fixed to the south effectively ensures that all eruptive events, regardless of wind direction, will produce a layer of contaminated ash at the receptor location. The use of the transition-phase BDCFs results in the assumption of high air-mass loading conditions above this contaminated ash layer indefinitely, and ensures that all radionuclides in the layer are available in the upper 1 cm for resuspension and inhalation. The actual thickness of layers of contaminated sediments that might be deposited at the location of the receptor group and the distribution of radionuclides within them is unknown, but radionuclide concentrations in sediment will be more dilute than those calculated for the initial ash fall. Thus, the combination of the transition-phase BDCFs and the assumption of a fixed wind direction reasonably compensate for uncertainty regarding surface redistribution processes by resulting in a calculated dose that is greater for all eruptive events than what could reasonably be expected from surface redistribution processes.

3.10.5 Results and Interpretation: Evaluation of Issues Important to Performance

Other than the BDCFs discussed in Section 3.9, the most important factors contributing to the expected annual doses calculated for the igneous disruption scenario are the concentrations of radionuclides reaching the critical group, both in groundwater and ash. For the intrusion pathway, concentrations of radionuclides in groundwater are calculated as part of the entire TSPA-SR analysis, and are described in Section 4.2. There are no intermediate results available from the igneous intrusion groundwater transport model other than the number of packages damaged, as discussed in the previous section. For the volcanic eruption pathway, however, ASHPLUME calculates the thickness of the ash layer at the location of the critical group and the mass of radionuclides per unit area in the ash layer. These results are discussed in the following sections.

3.10.5.1 Ash Layer Thickness 20 Kilometers from the Potential Repository

The thickness of the ash layer that could be deposited 20 km from the potential repository is uncertain. Figure 3.10-14 shows two sets of ASHPLUME results from 300 realizations using sampled values for the parameters listed in Table 3.10-4. One set of results shows a distribution of ash layer thickness at 20 km calculated assuming that the wind always blows directly south, toward the critical group. The other set of results includes a sampling of wind directions, so that wind blows away from the critical group for a significant number of realizations. Sampling of all input parameters except wind direction is the same in each set of realizations.

Figure 3.10-14 shows that the maximum thickness of the calculated ash layer at 20 km is relatively insensitive to the wind direction. This is reasonable, and given a large enough sample size, the two curves ought to converge at the upper end of the distribution. The median calculated thickness (and all other thickness below the upper bound) at any one specified location are strongly sensitive to wind direction. This is also a reasonable observation. If the wind blows in any direction other than directly at the location of interest, the ash layer thickness will be thinner. The fixed wind-direction curve can be thought of as representing the thickness of the ash layer at 20 km at the midpoint of the plume, regardless of wind direction. At any location off the midpoint of the plume, layer thickness will be less.

Overall, the calculated thickness appears reasonable. The median eruptive event produces an ash layer less than 1 cm thick, 20 km downwind. The minimum ash layer calculated for the midpoint of the plume at 20 km is less than 0.1 mm, corresponding to a relatively small eruption that produces only a dusting of ash at that distance. The maximum ash layer is 36 cm, corresponding to a large eruption that produces a major ash fall covering a large area. There is no field evidence suggesting that basaltic eruptions in the past in the Yucca Mountain region have produced ash falls this thick at 20 km, and the calculation appears, therefore, to provide a reasonable bound for uncertainty in the magnitude of future eruptions.

3.10.5.2 Waste Concentrations in Ash 20 Kilometers from the Repository

Figure 3.10-15 shows calculated concentrations of waste (i.e., radioactive material) on the ground surface 20 km from the potential repository. As in Figure 3.10-14, this figure shows results of two sets of 300 realizations, realizations using sampled values for the parameters listed in Table 3.10-4. One set of results shows a distribution of waste concentrations at 20 km, calculated assuming that the wind always blows directly south, toward the critical group. The other set of results includes a sampling of wind directions, so wind blows away from the critical group for a significant number of realizations. Sampling of all input parameters except wind direction is the same in each set of realizations.

Results are consistent with those shown for ash layer thickness in Figure 3.10-14. The maximum concentrations are relatively insensitive to wind direction, but the median (and all other concentrations below the upper bound) are strongly sensitive to the direction of the wind at the time of the eruption. The median concentration of waste at the midpoint of the plume 20 km from the potential repository is somewhat more than 10^{-6} g/cm^2, distributed in an ash layer less than 1 cm thick. Unlike ash layer thickness, there are no natural analogs to compare these results to, however simple hand calculations provide a crude check on their reasonableness. A surface concentration of 10^{-6} g/cm^2 corresponds, for example, to the concentration that would result from spreading the entire contents of a single waste package (approximately 10^7 g of radioactive material) uniformly over 1000 km^2 (32×32 km, or 10^{13} cm^2). The comparison is admittedly overly simplistic, because calculated concentrations involve complex, nonuniform distributions of ash and waste throughout a plume of variable size, and waste concentrations are likely to be higher closer to the point of eruption. Eruptions may affect substantially more than one package (see Table 3.10-4), but not all waste in the damaged packages will be entrained in the eruption. Overall, the ASHPLUME (CRWMS M&O 1999 [150744]) calculations appear to be providing a reasonable distribution of possible surface concentrations of waste 20 km from the point of eruption.

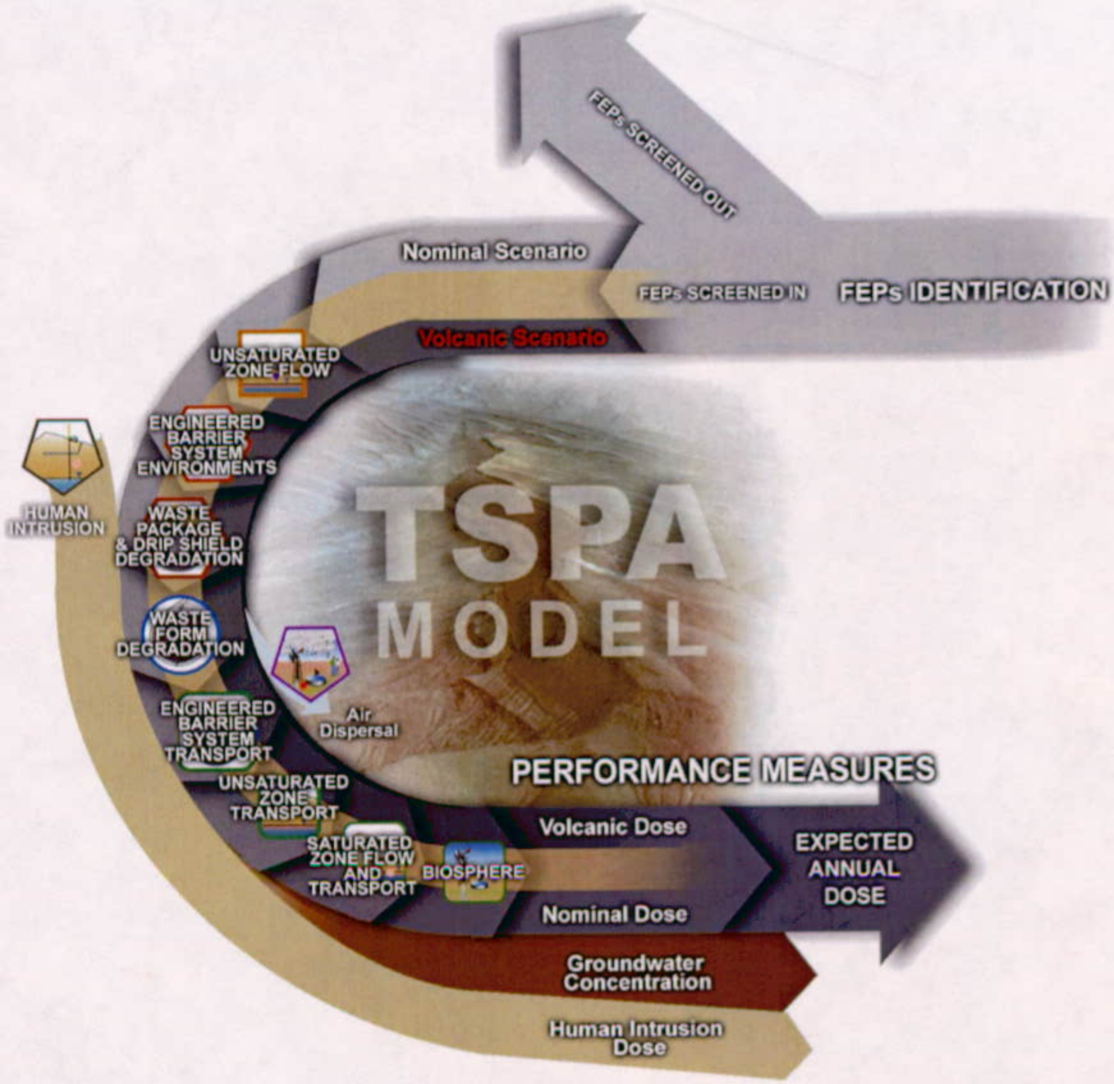

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Figure 3.1-1. Summary of the Total System Performance Assessment-Site Recommendation Scenarios, Models and Analyses

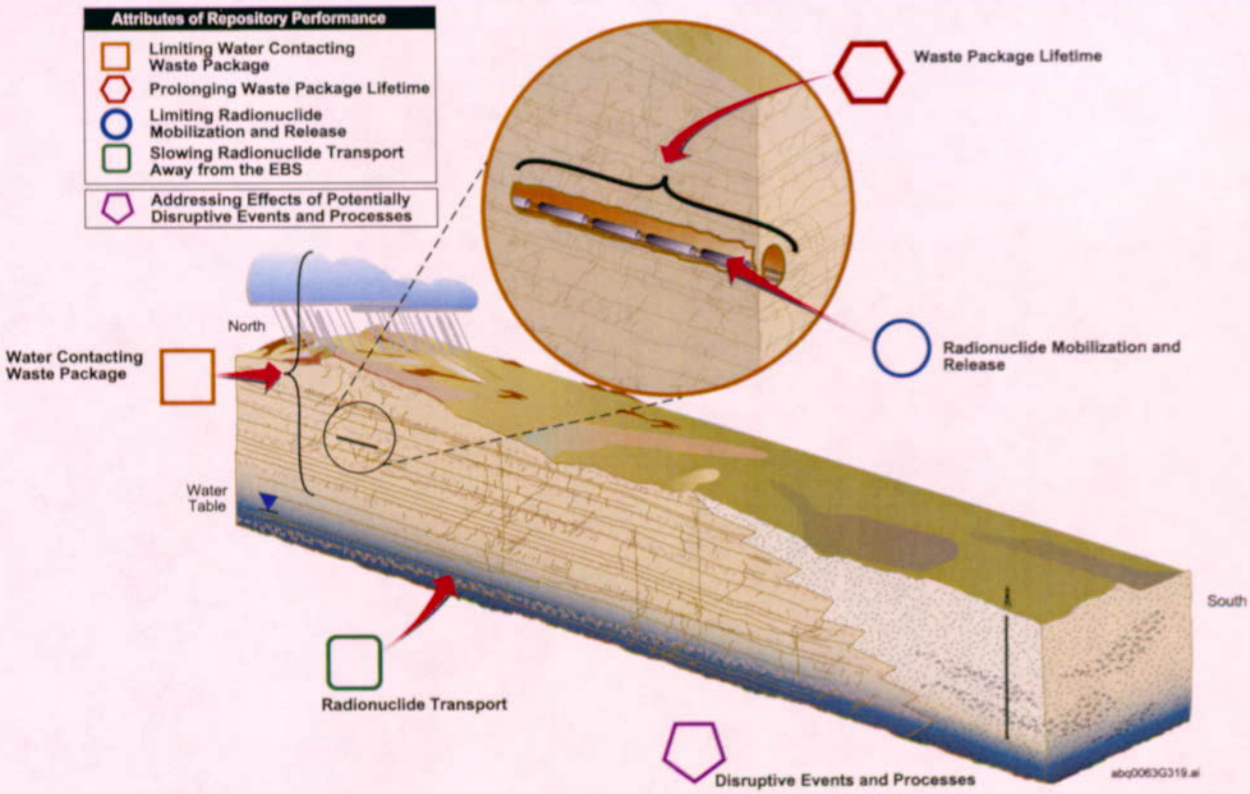

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Figure 3.1-2. Attributes of Repository Performance Included in Total System Performance Assessment-Site Recommendation

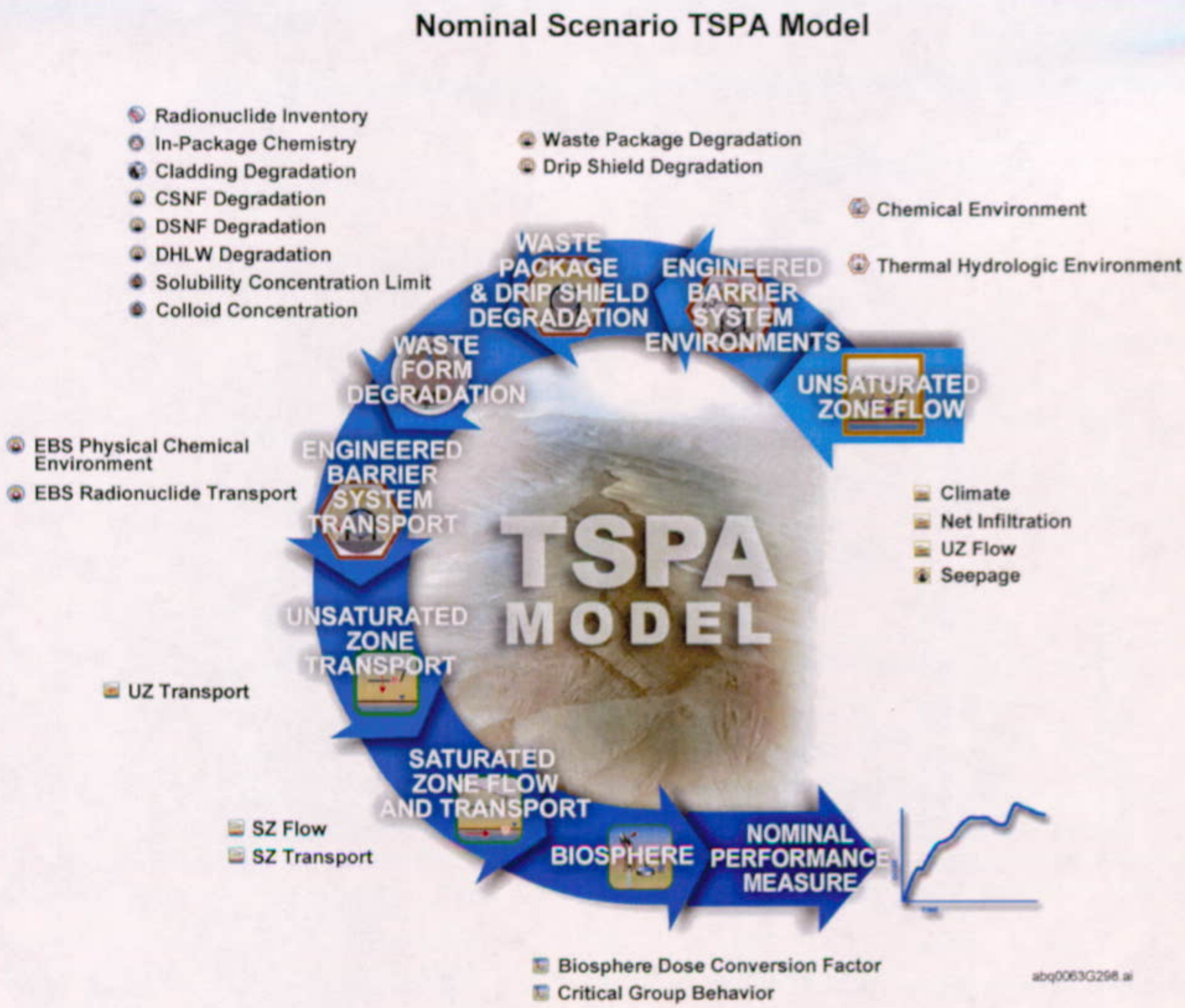

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Figure 3.1-3. Process Model Factors Included in Total System Performance Assessment-Site Recommendation

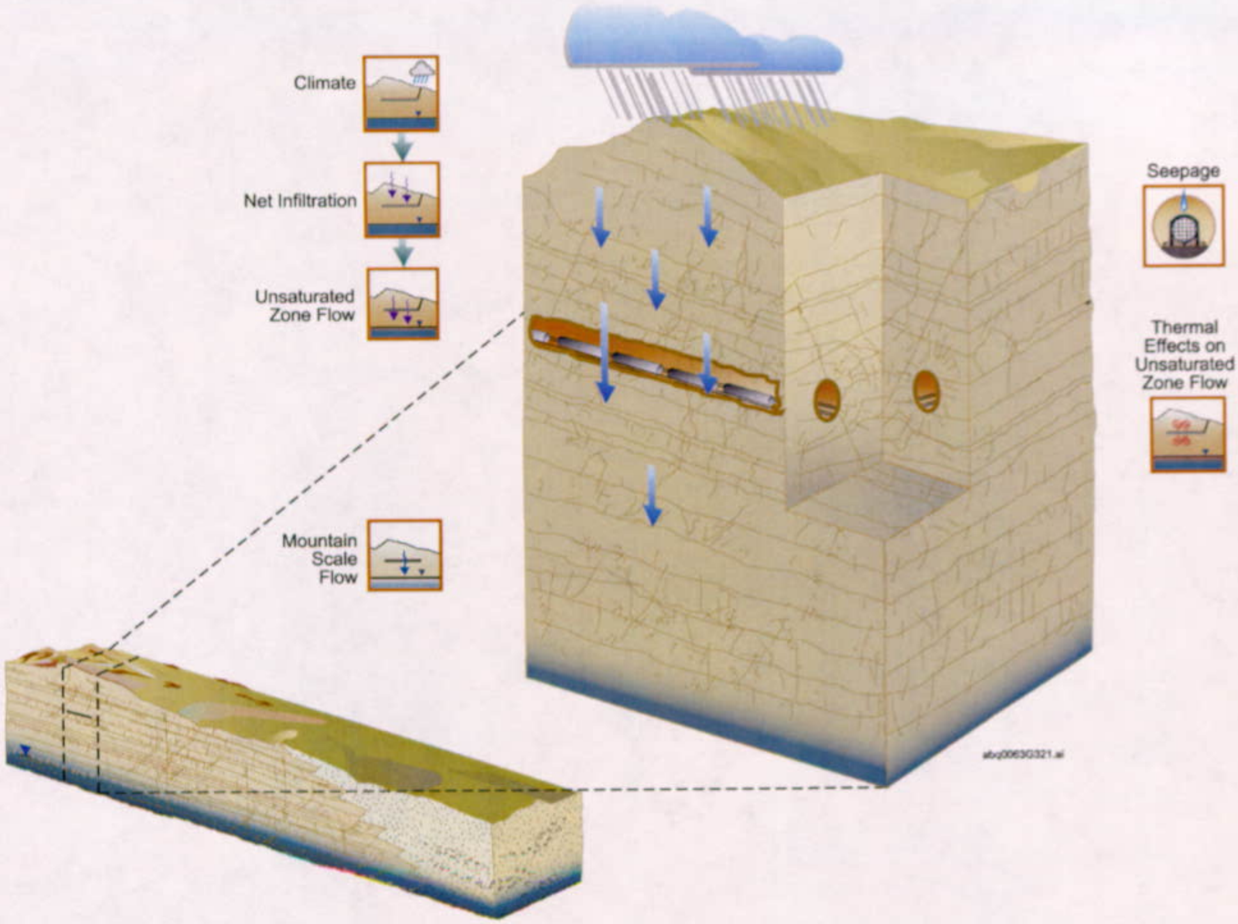

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Figure 3.1-4. Process Model Factors Affecting Water Contacting Waste Packages Included in the Total System Performance Assessment-Site Recommendation

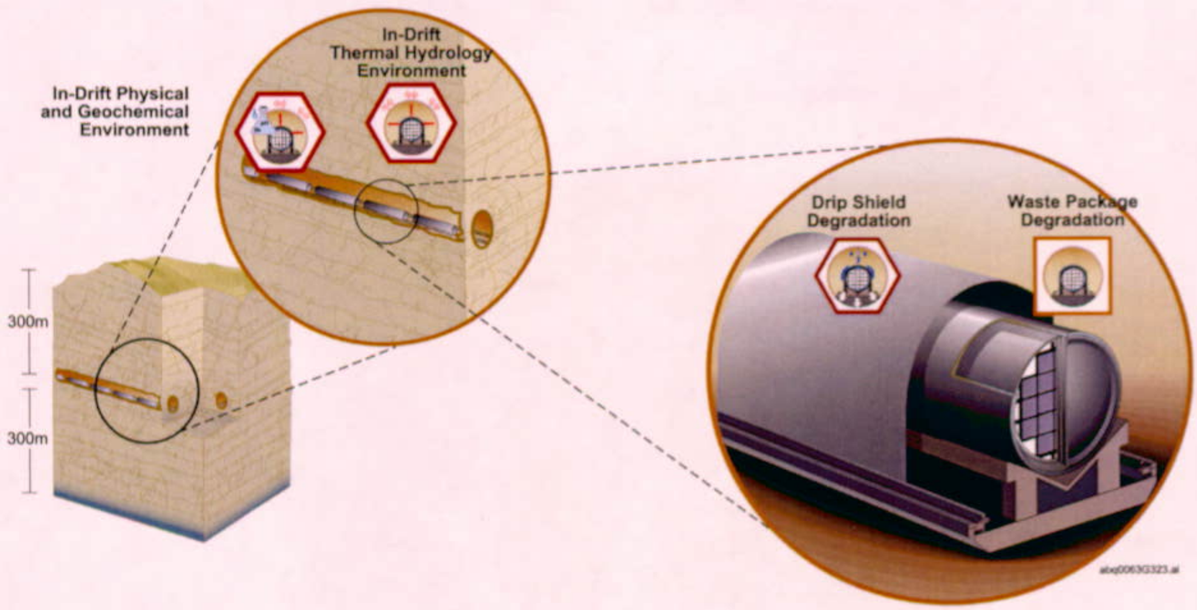

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Figure 3.1-5. Process Model Factors Affecting Waste Package Lifetime Included in the Total System Performance Assessment-Site Recommendation

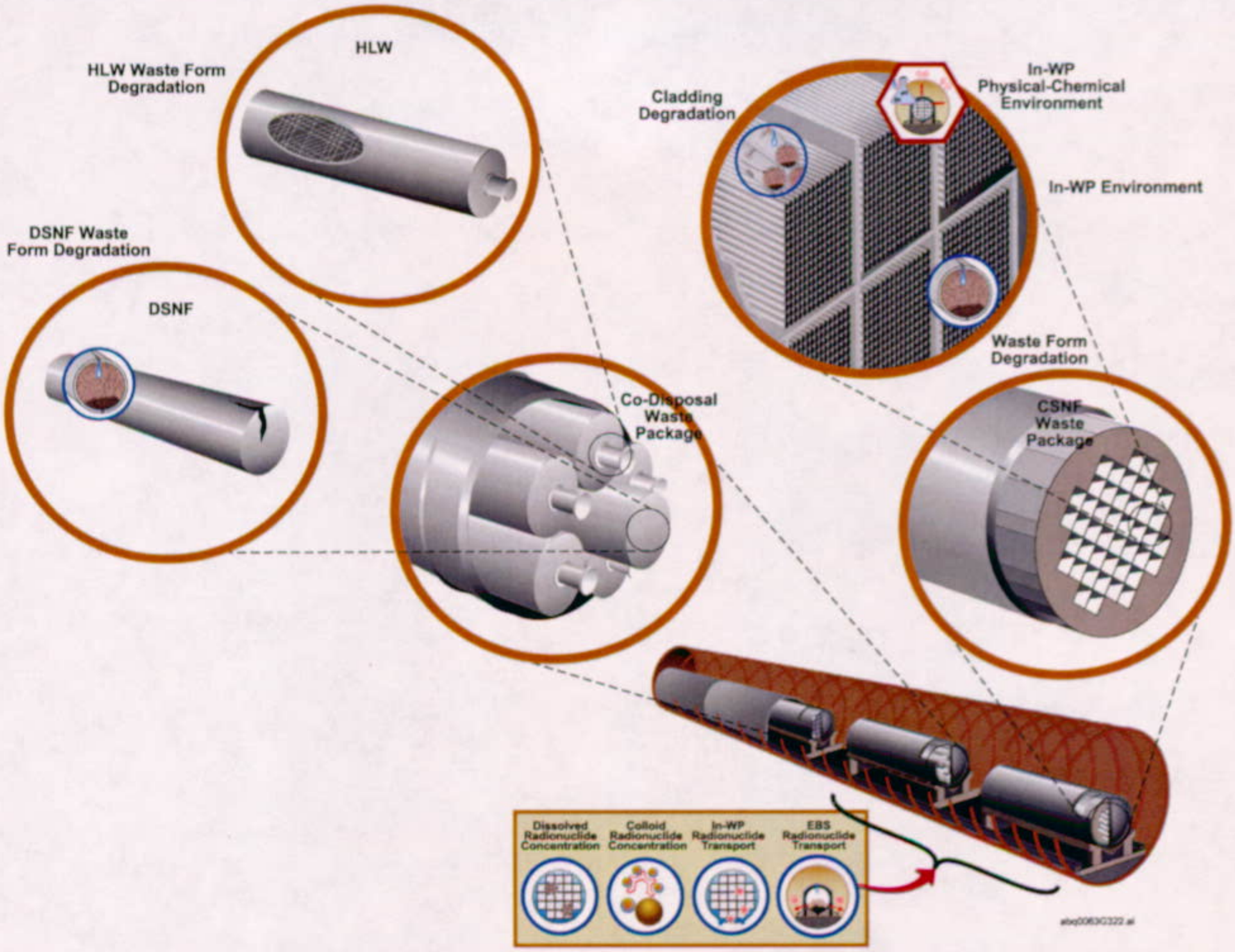

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Figure 3.1-6. Process Model Factors Affecting Radionuclide Mobilization and Release from the Engineered Barriers Included in the Total System Performance Assessment-Site Recommendation

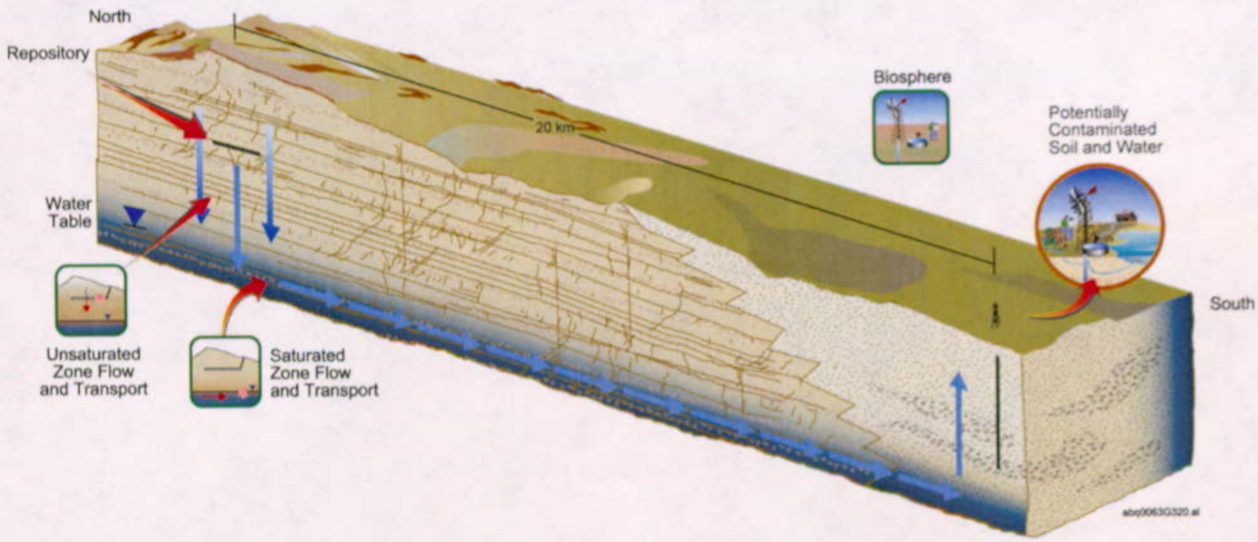

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Figure 3.1-7. Process Model Factors Affecting Radionuclide Transport Included in Total System Performance Assessment-Site Recommendation

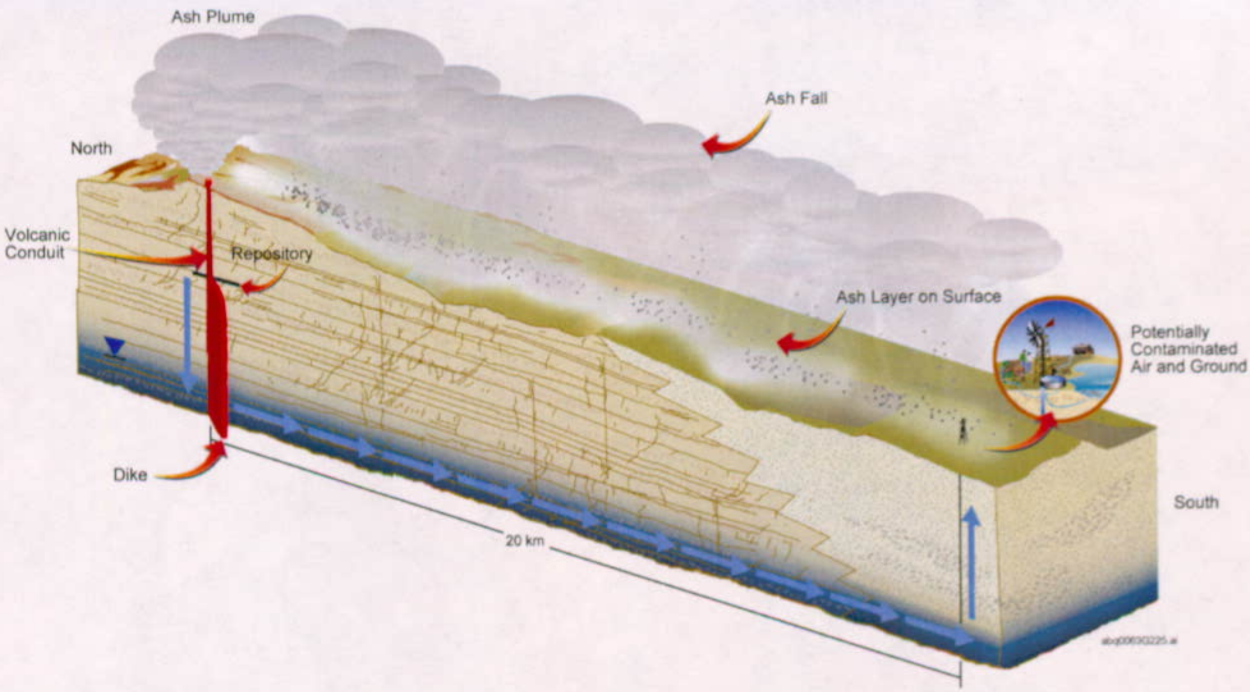

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Figure 3.1-8. Process Model Factors Affecting Probability and Consequences of Disruptive Events Included in the Total System Performance Assessment-Site Recommendation

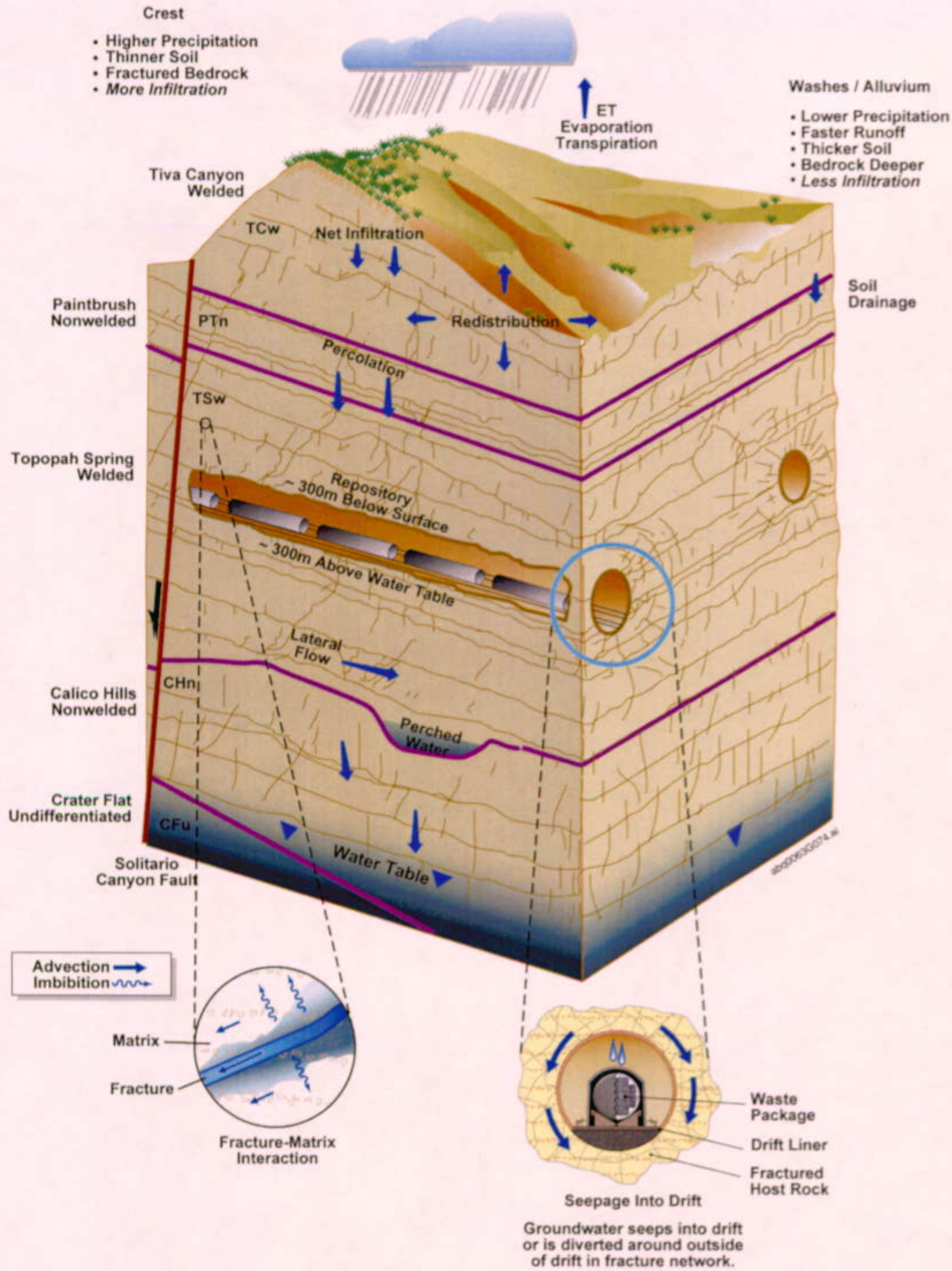

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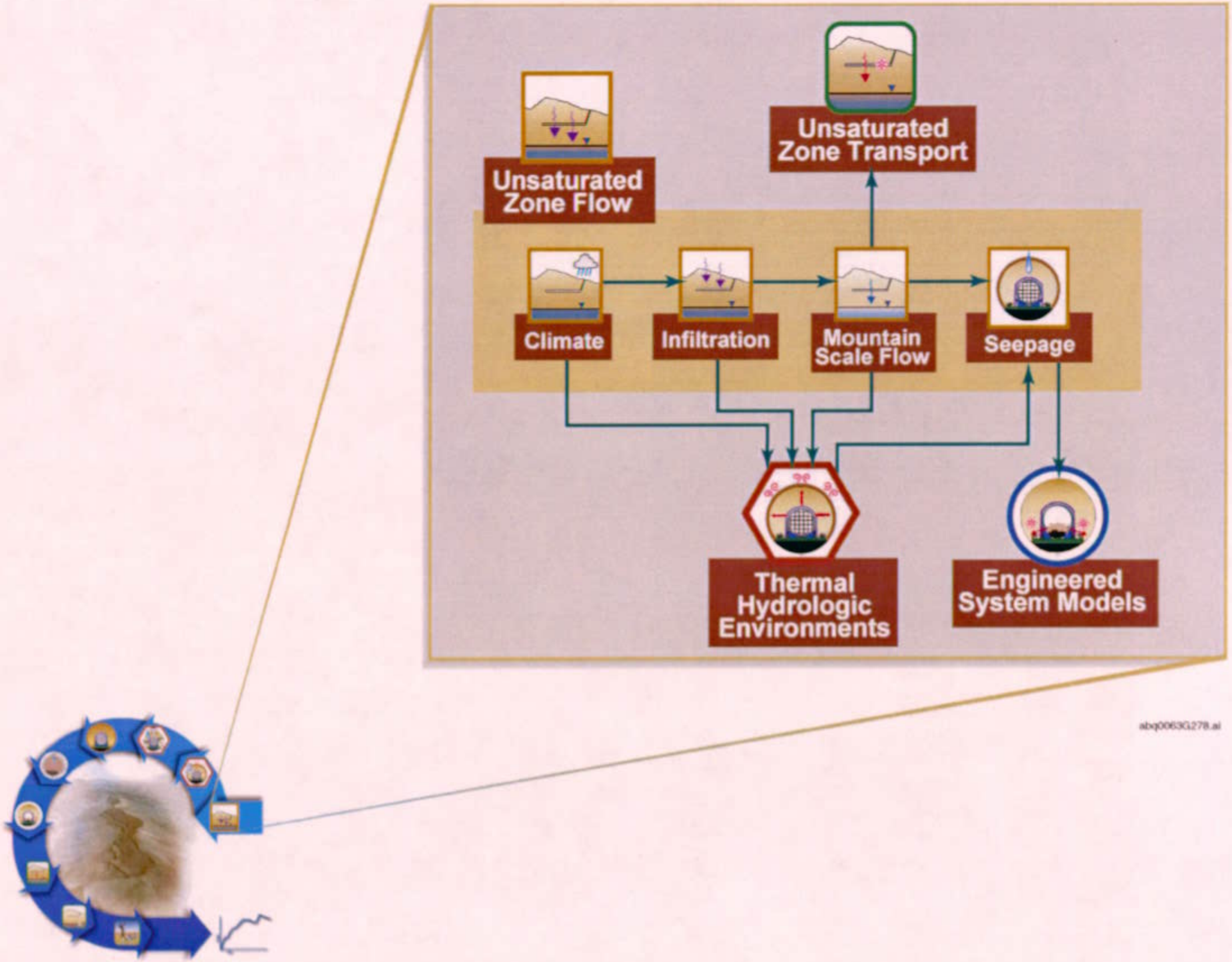
Figure 3.2-1. Conceptual Drawing of Unsaturated Zone Flow Processes at Different Scales



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NOTE: Index figure in lower left is same as Figure 2.1-6

Figure 3.2-2. Information Flow Diagram for Unsaturated Zone Flow

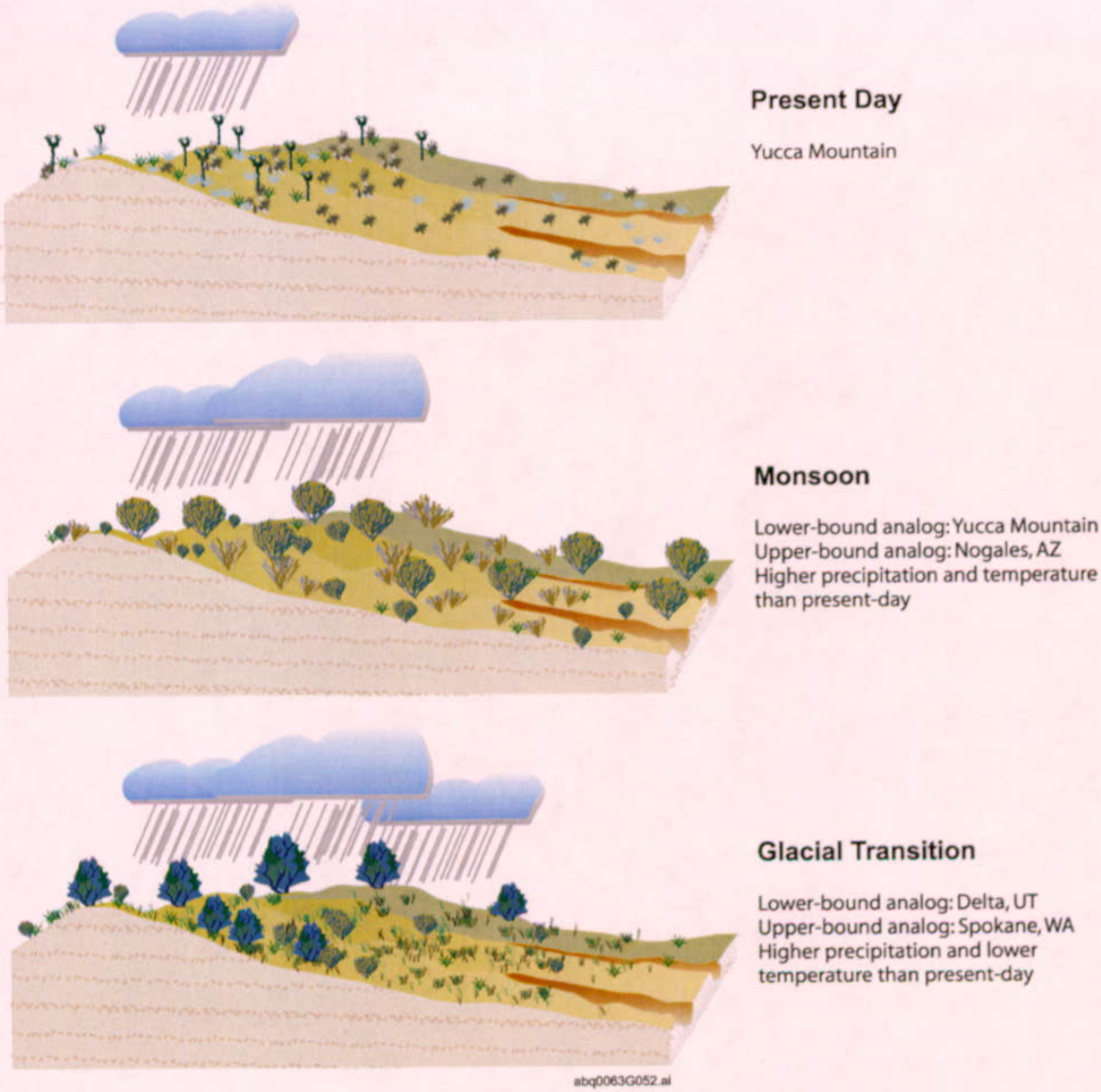

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Figure 3.2-3. Conceptual Drawing of Projected Climates for Yucca Mountain

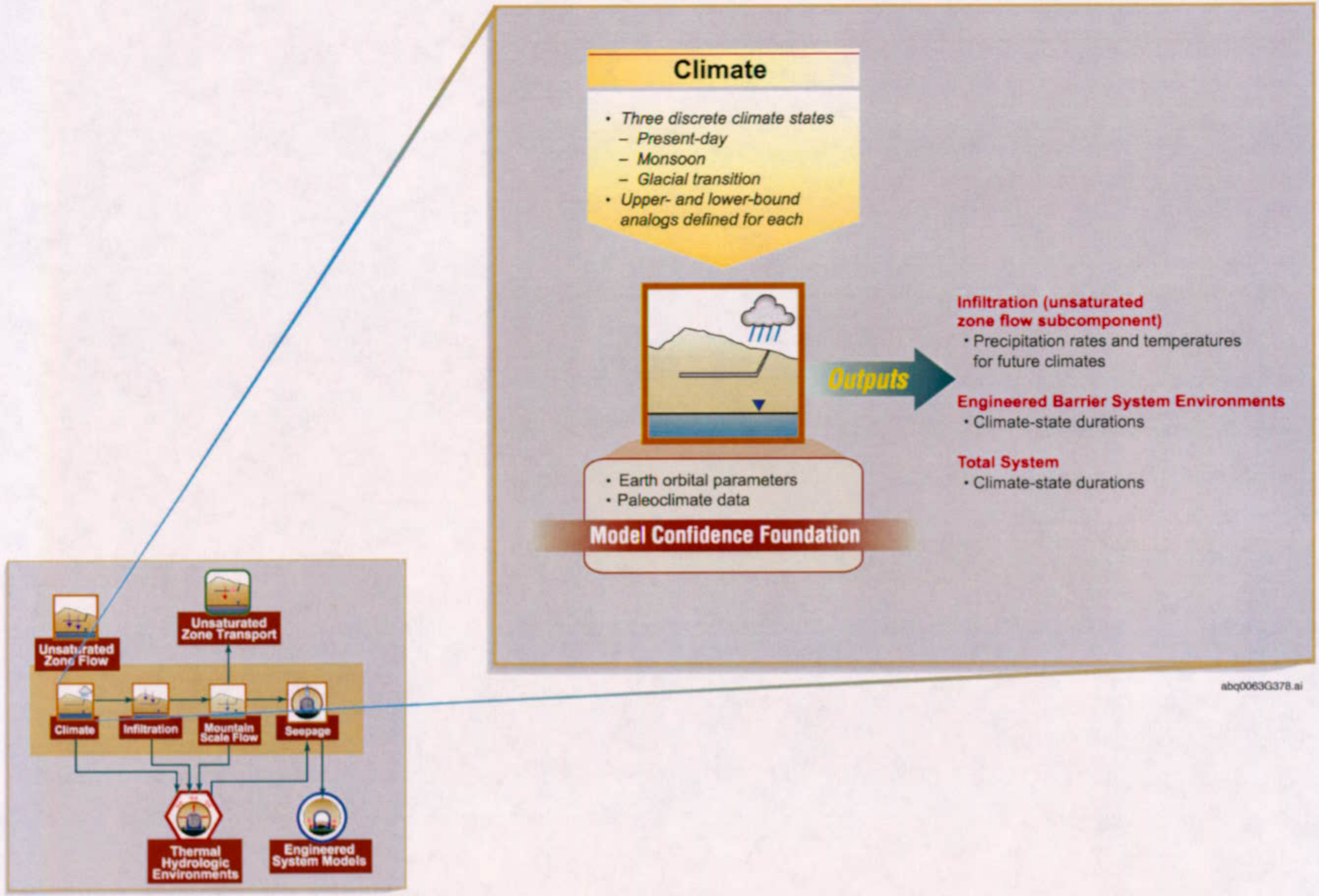

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Figure 3.2-4. Connections between Climate and Other Total System Performance Assessment Model Components

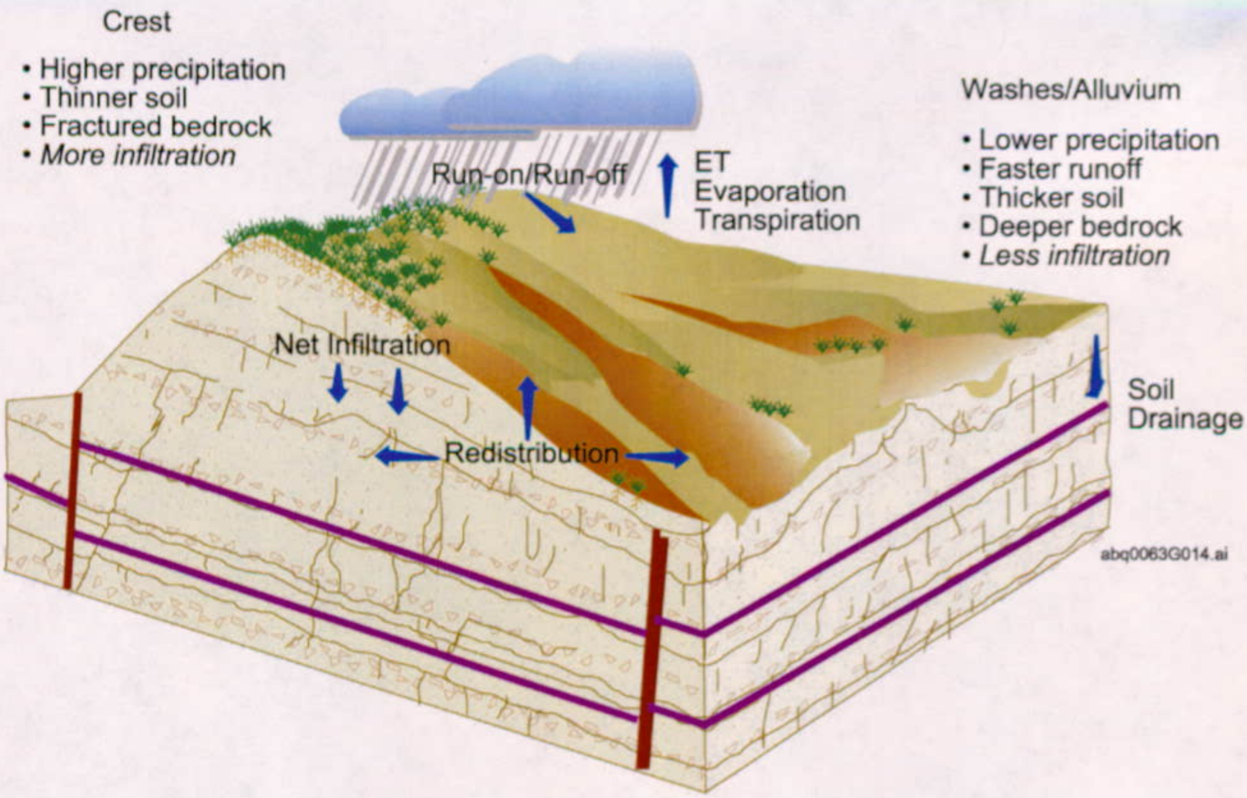

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Figure 3.2-5. Conceptual Drawing of Infiltration Processes

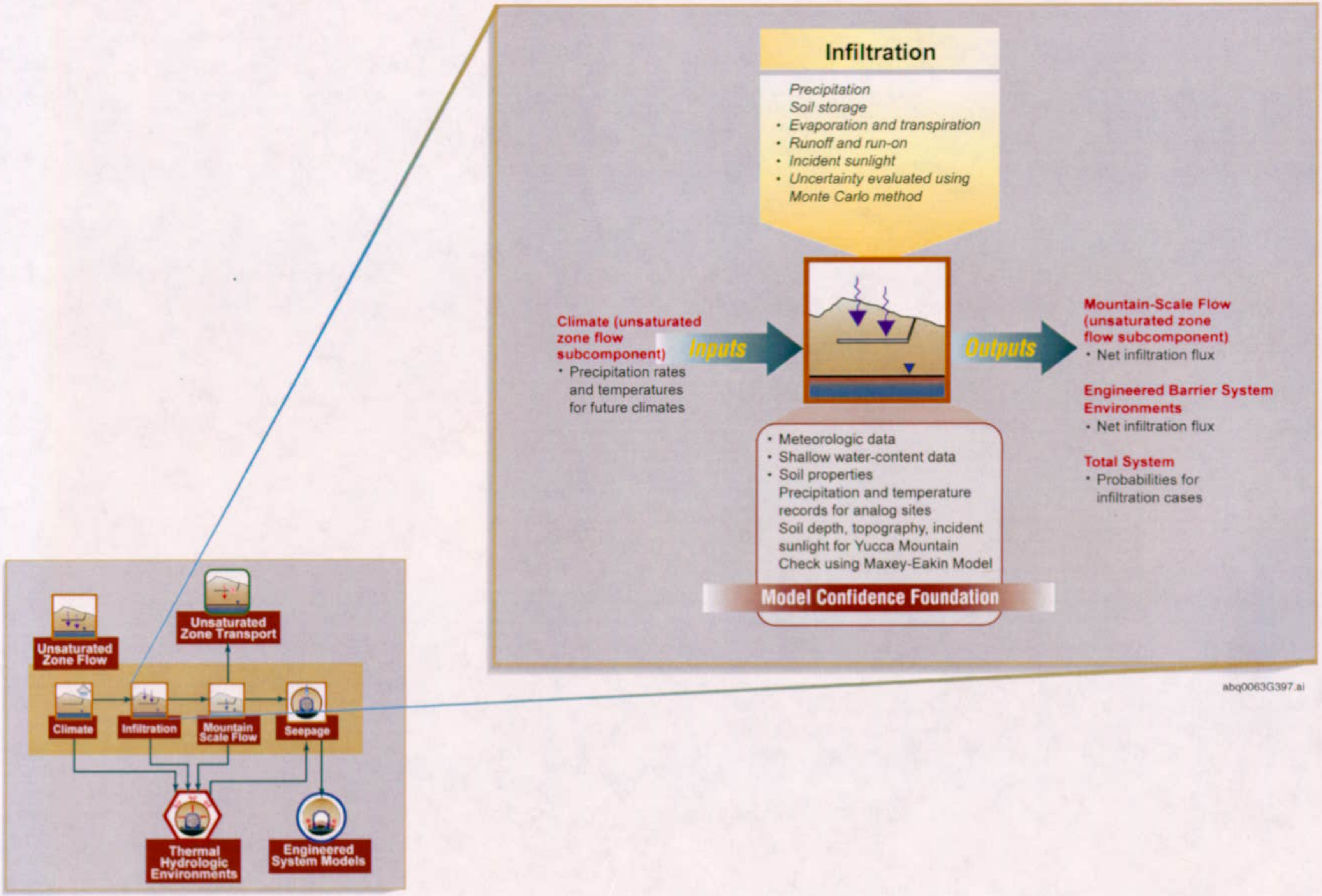

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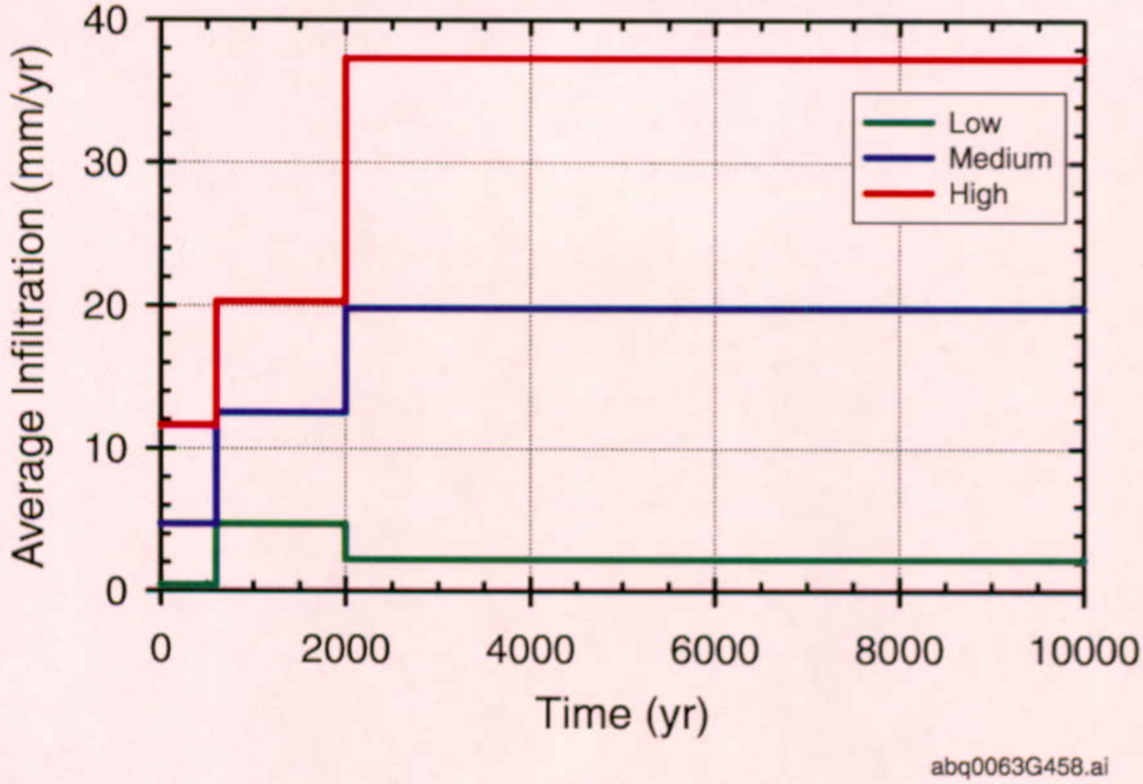
Figure 3.2-6. Connections Between Infiltration and Other Total System Performance Assessment Model Components



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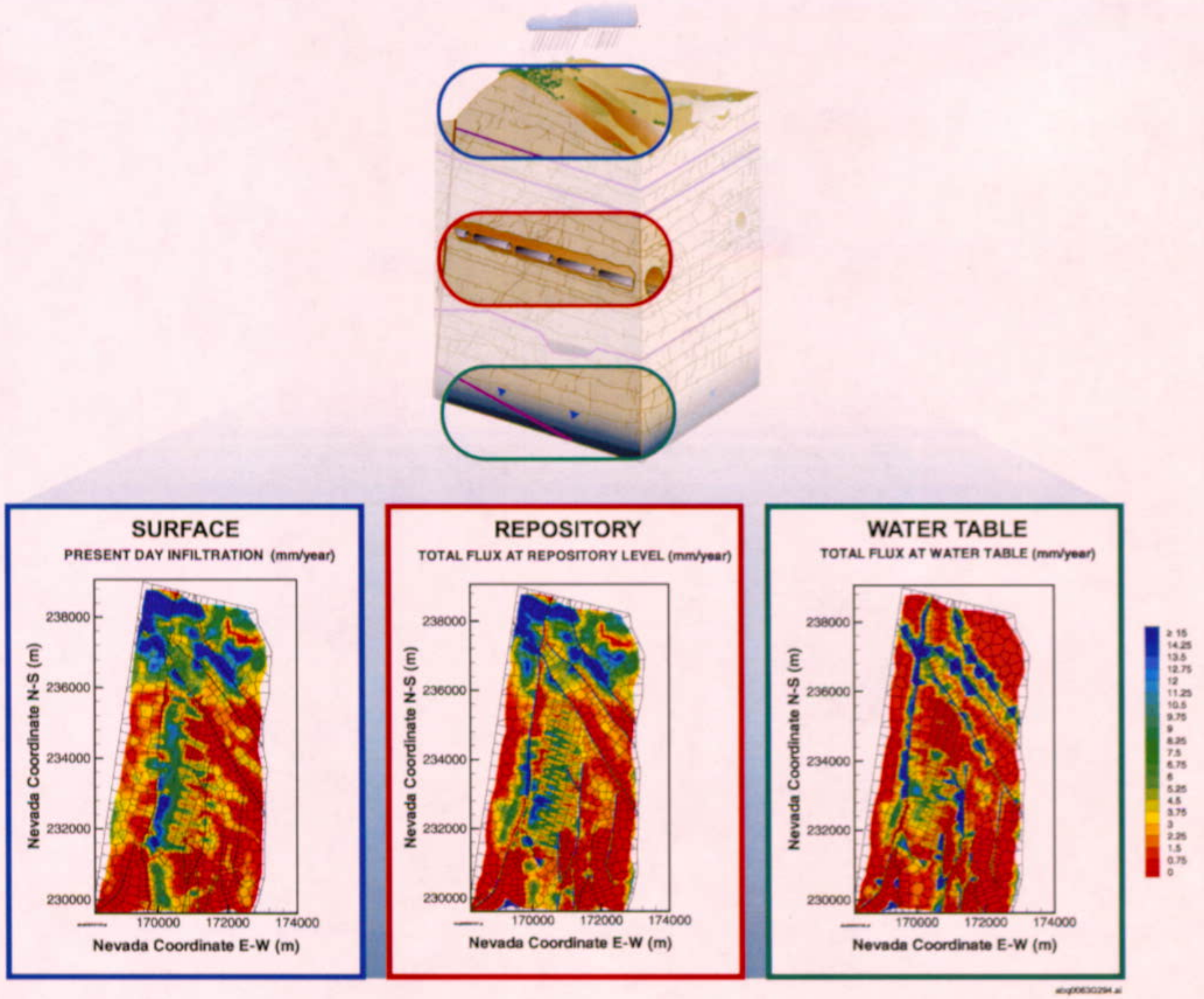
Source: Data from USGS 2000 [123650], Tables 6-10, 6-14, and 6-19

Figure 3.2-7. Repository-Average Net Infiltration for the Three Infiltration Cases



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Source: Adapted from CRWMS M&O 2000 [145774], Figures 3.7-4a, 3.7-10a, and 3.7-10b

Figure 3.2-8. Total Percolation Flux at Three Depths

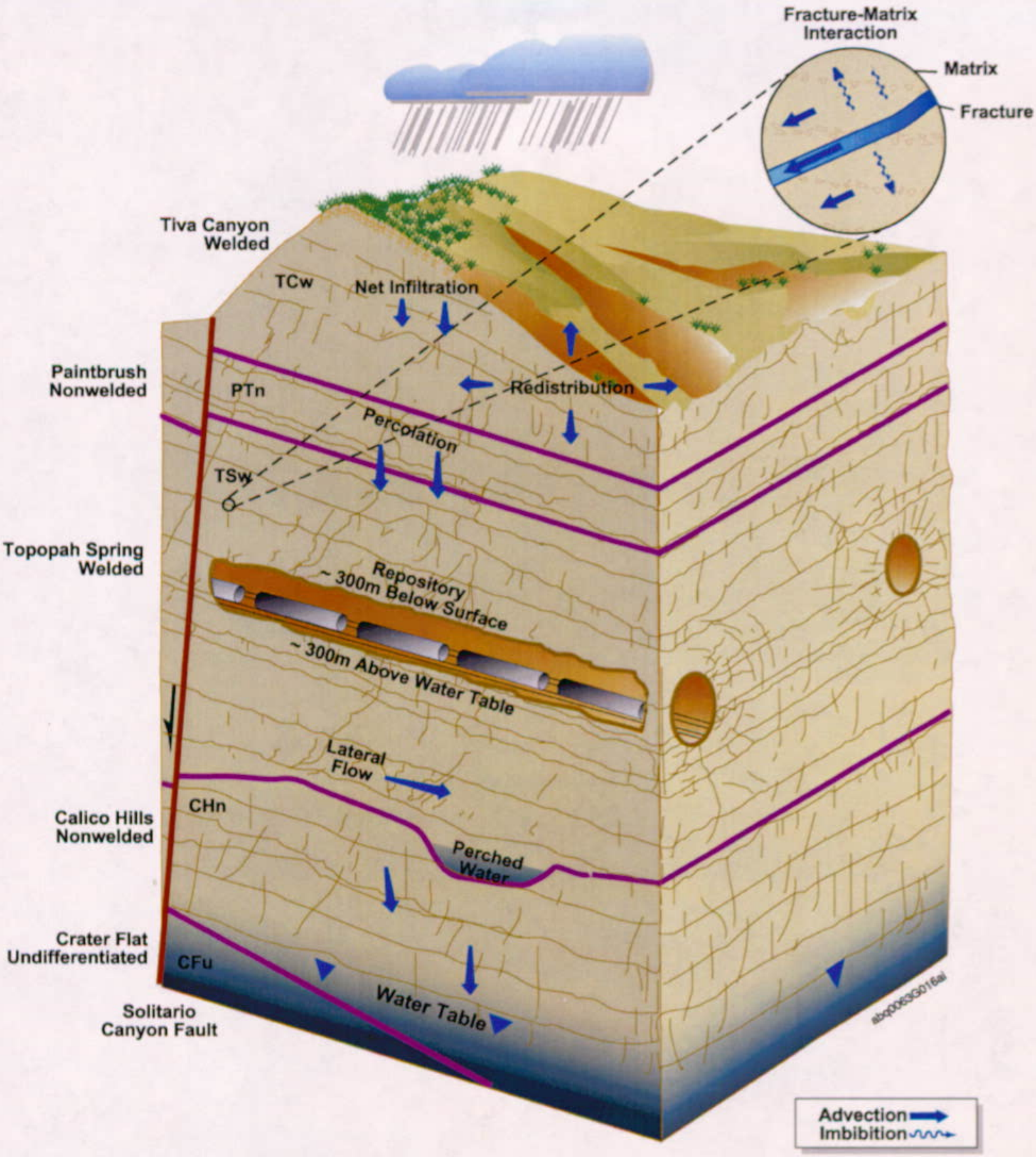

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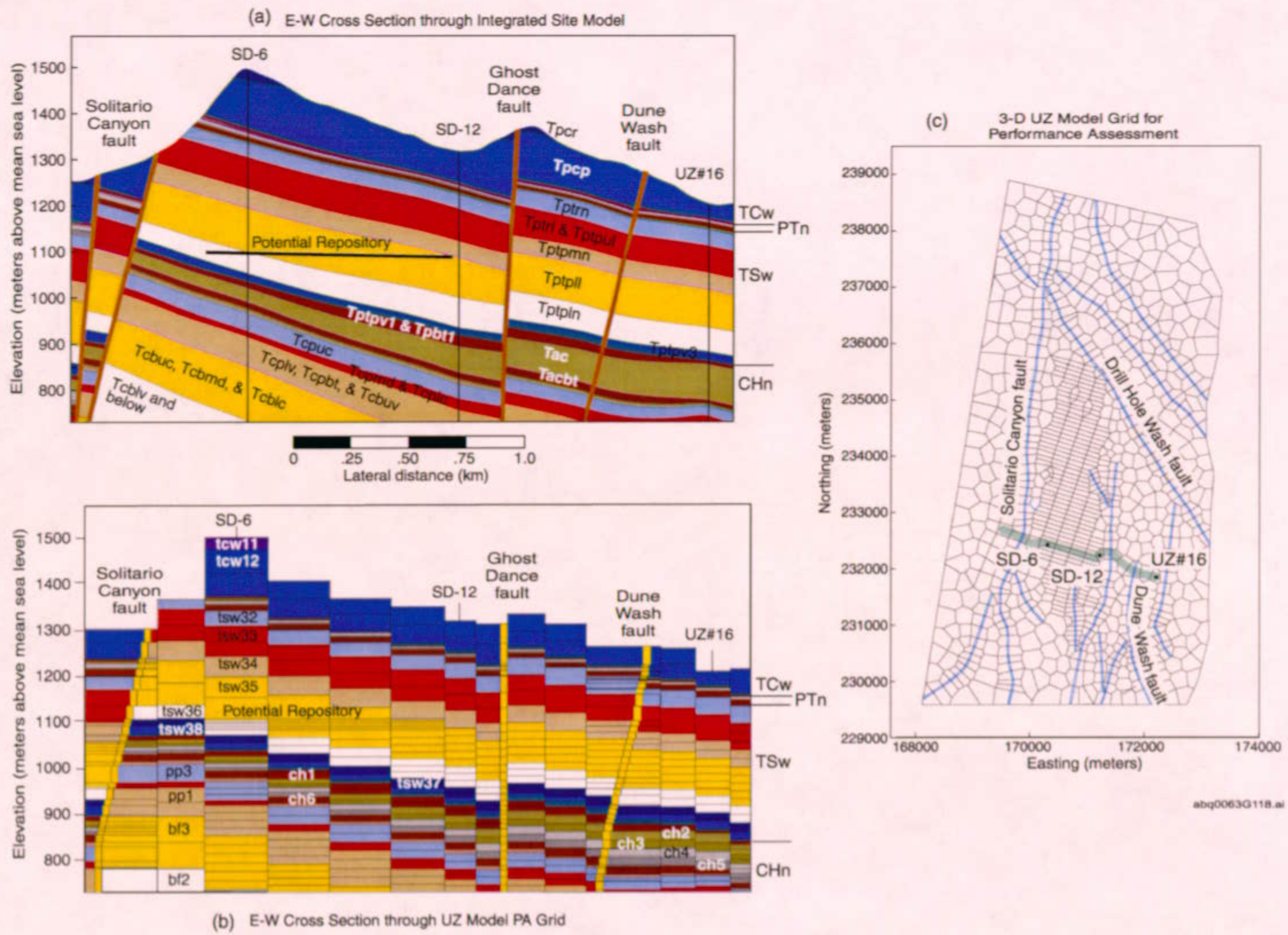
Figure 3.2-9. Conceptual Drawing of Mountain-Scale Flow Processes



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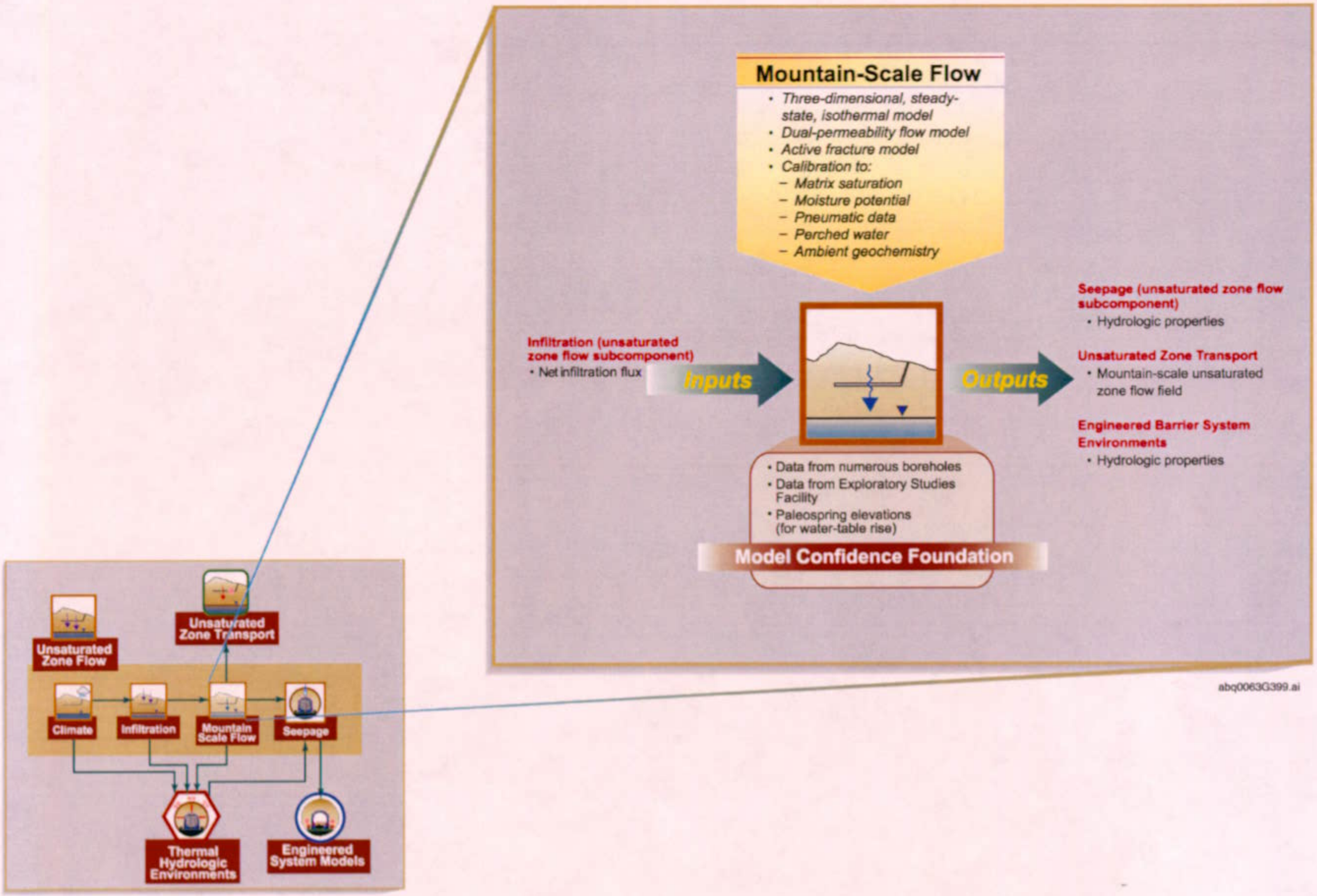
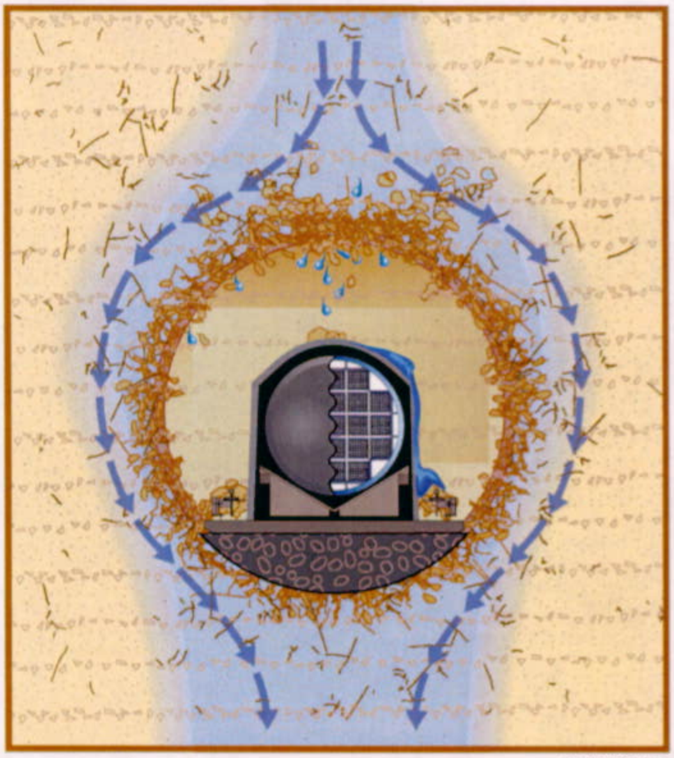
Source: Adapted from CRWMS M&O 2000 [145774], Figure 3.4-8

Figure 3.2-10. Stratigraphy and Mesh for Mountain-Scale Flow Model



abq0063G399

Figure 3.2-11. Connections Between Mountain-Scale Flow and Other Total System Performance Assessment Model Components

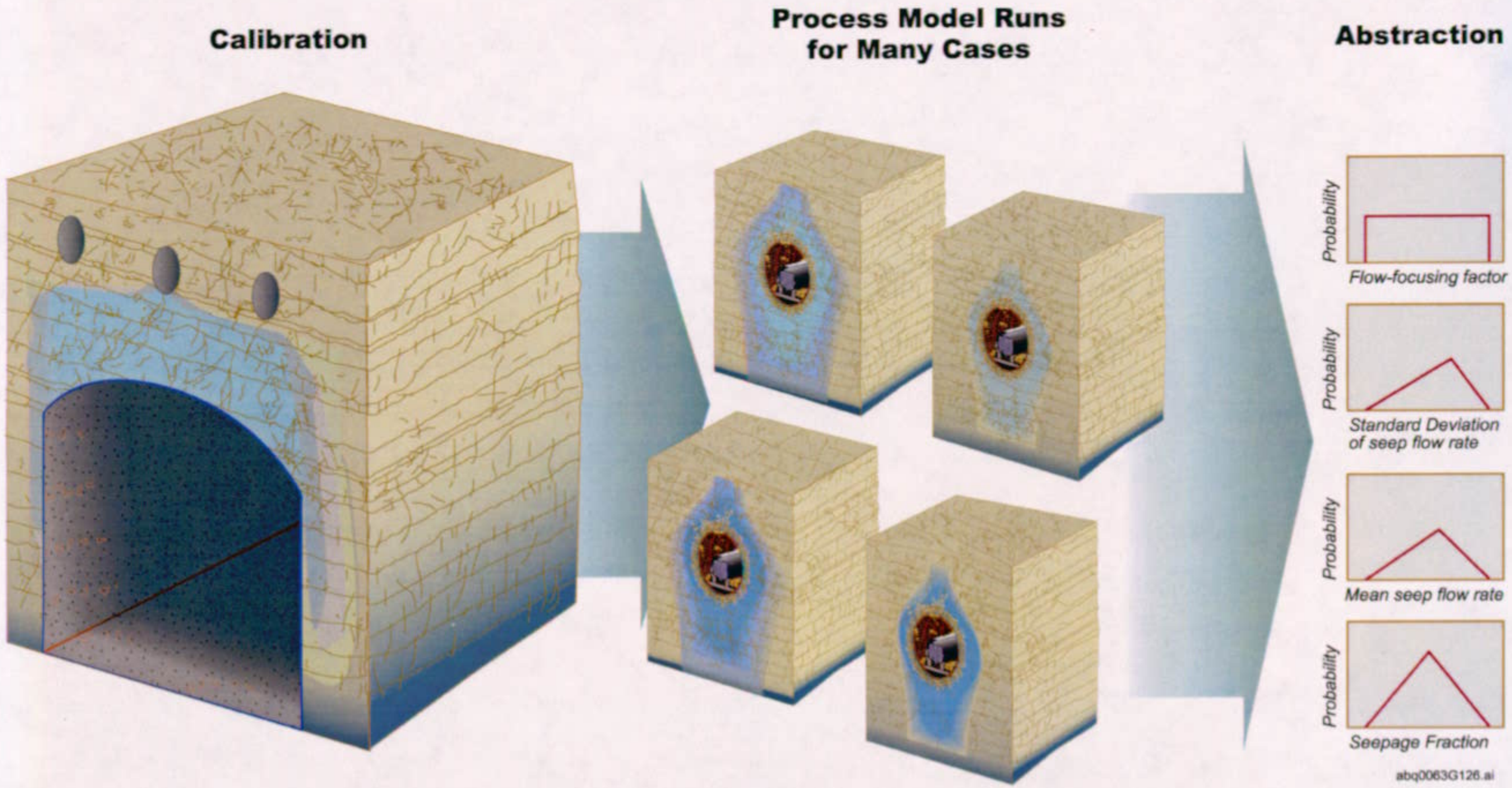


Important Factors for Seepage:

- Drift acts as capillary barrier (water is diverted around drift)
- Fracture permeability and capillarity
- Heterogeneity of hydrologic properties
- Channeling of flow
- Drift geometry and degradation
- Excavation-disturbed zone
- Thermal perturbation from waste

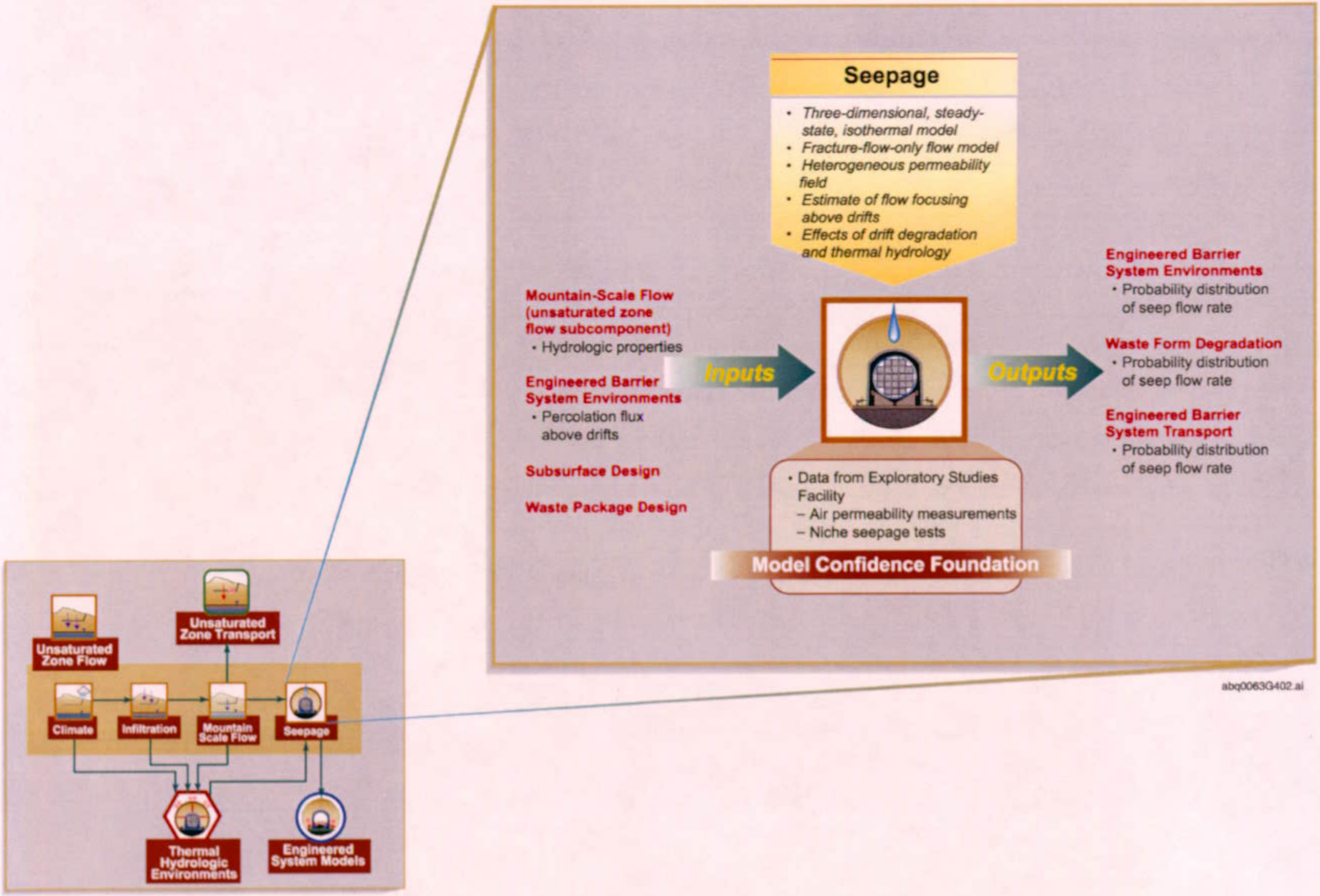
abq0063G018

Figure 3.2-12. Conceptual Drawing of Seepage Processes



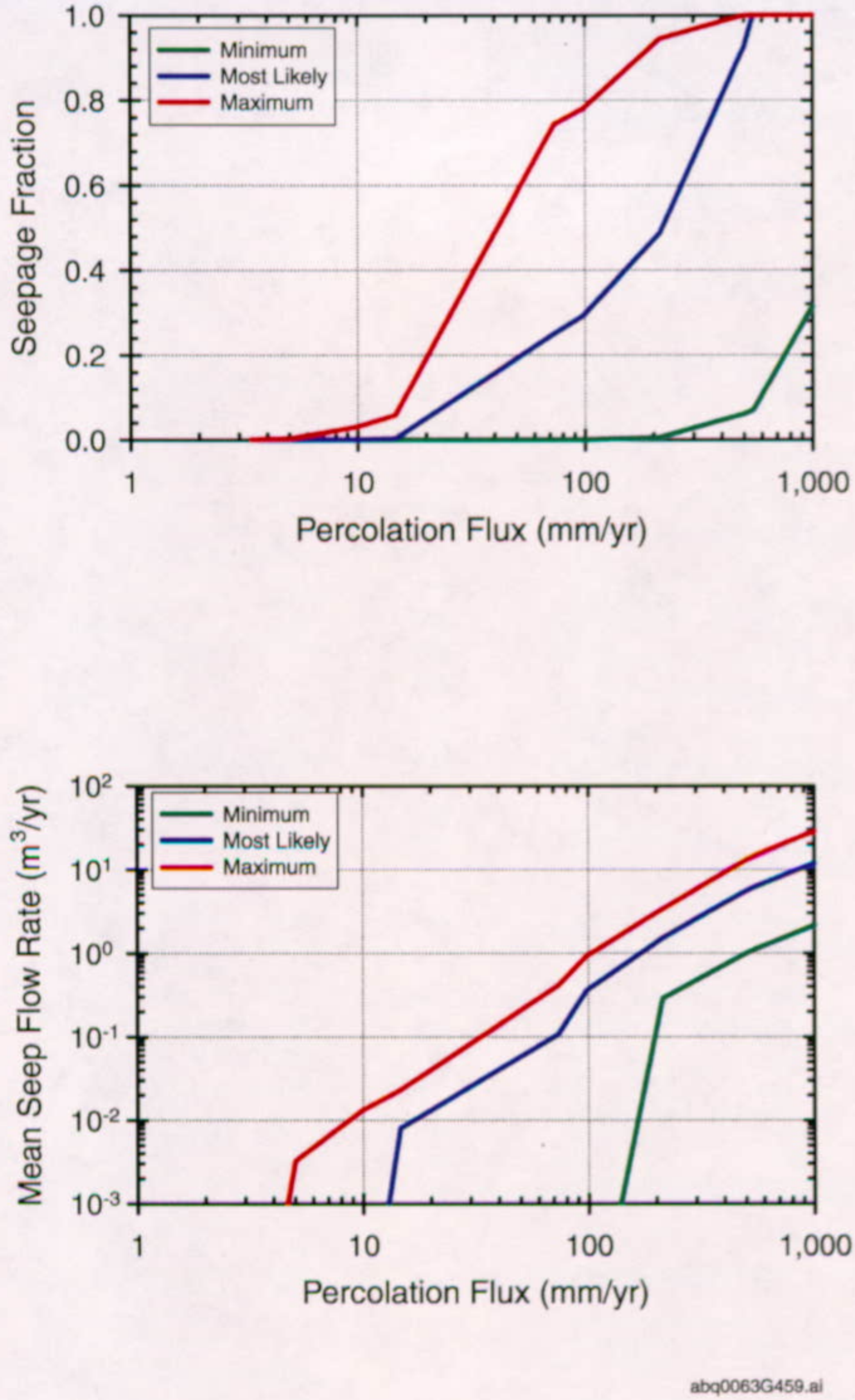
abq0063G126

Figure 3.2-13. Three-Step Process for Modeling Seepage into Drifts



abq0063G402

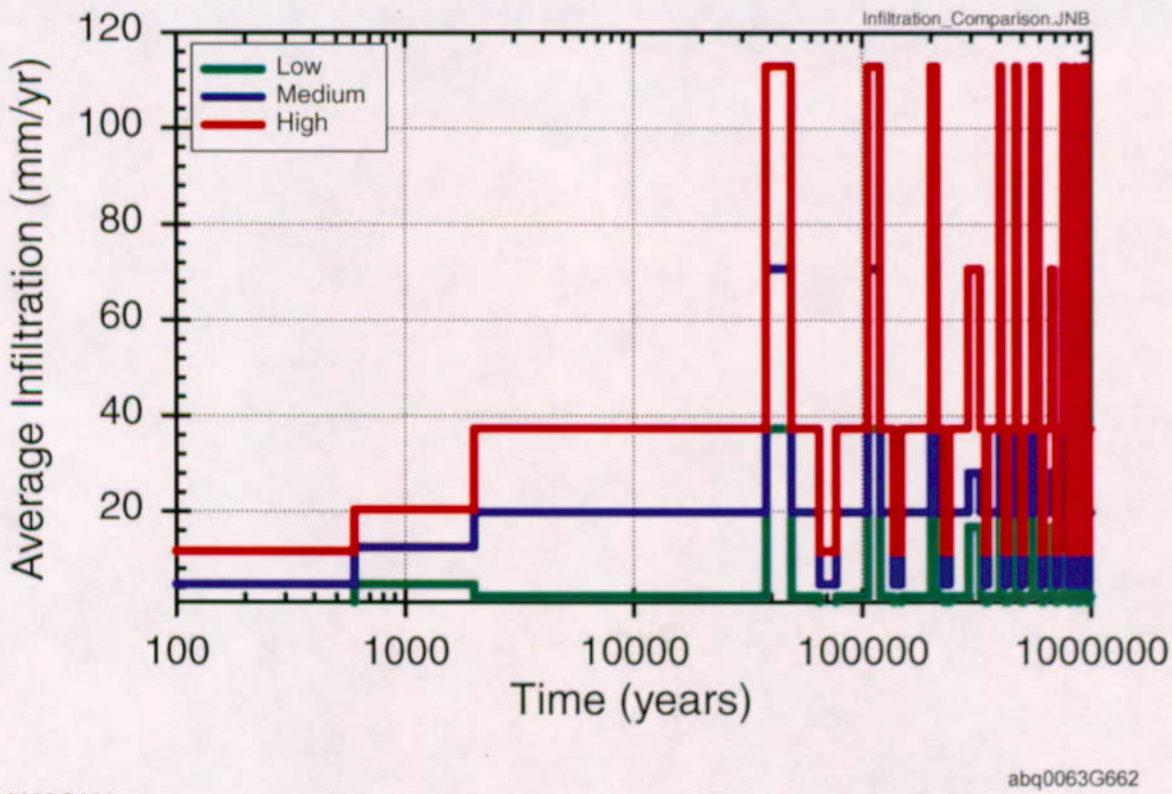
Figure 3.2-14. Connections Between Seepage into Drifts and Other Total System Performance Assessment Model Components



abq0063G459

Source: Adapted from CRWMS M&O 2000 [142004], Figures 2 and 3

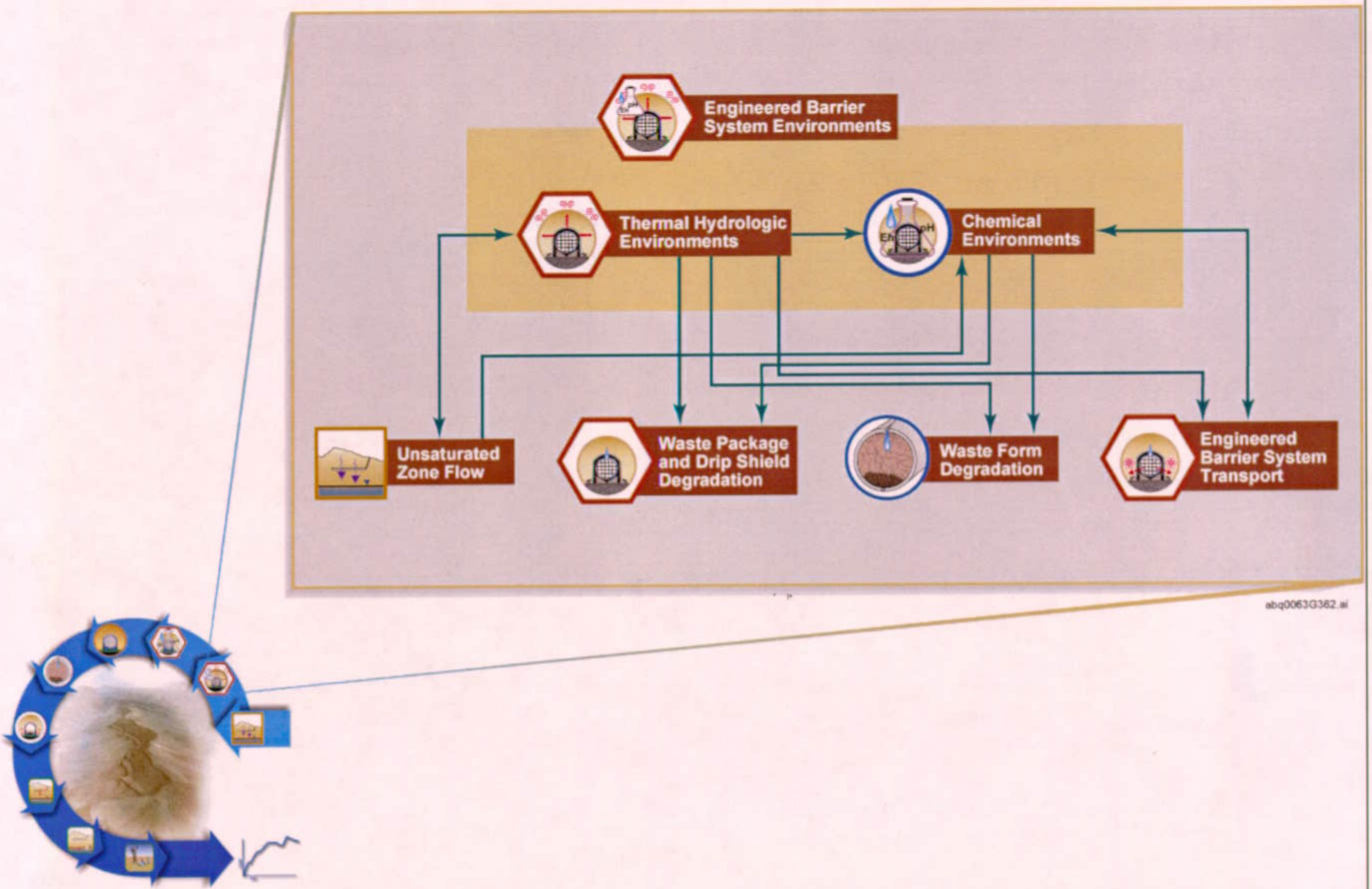
Figure 3.2-15. Uncertainty Distributions for Seepage Fraction and Mean Seep Flow Rate



abq0063G662

Source: Data from USGS 2000 [123650], Tables 6-10, 6-14, and 6-19; CRWMS M&O 2000 [153002]; Table 6-6

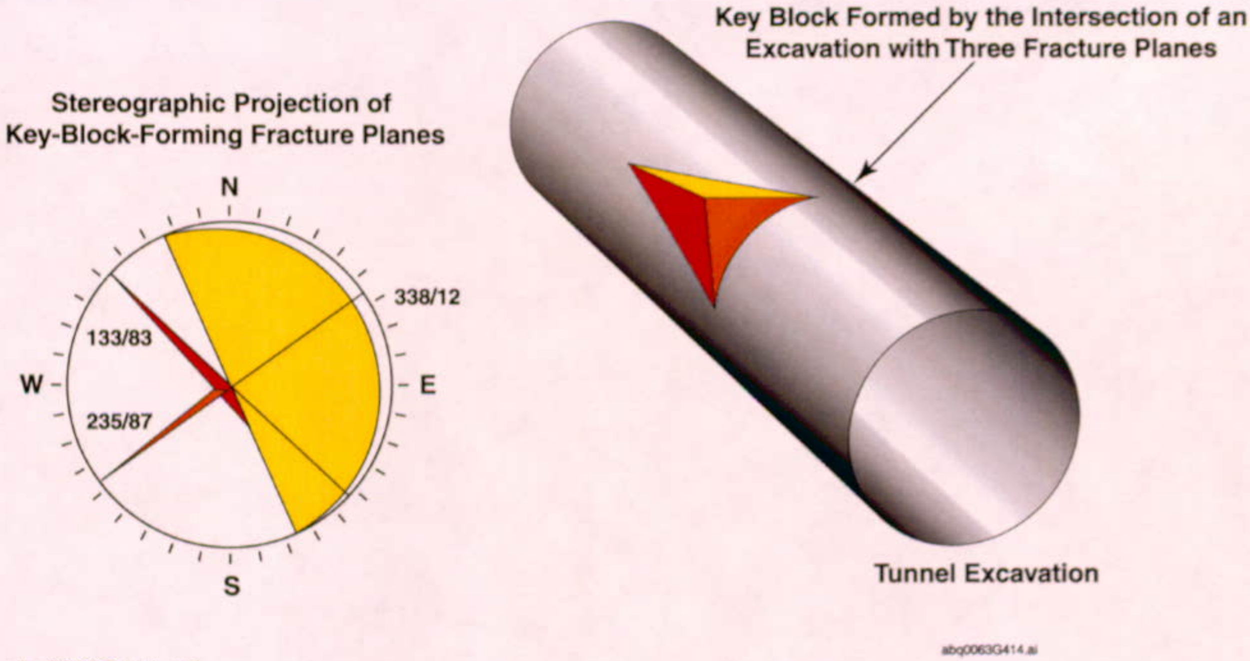
Figure 3.2-16. Repository-Average Net Infiltration over Time for the Alternative Climate Sequence



abq0063G362

NOTE: Index figure in lower left is same as Figure 2.1-6

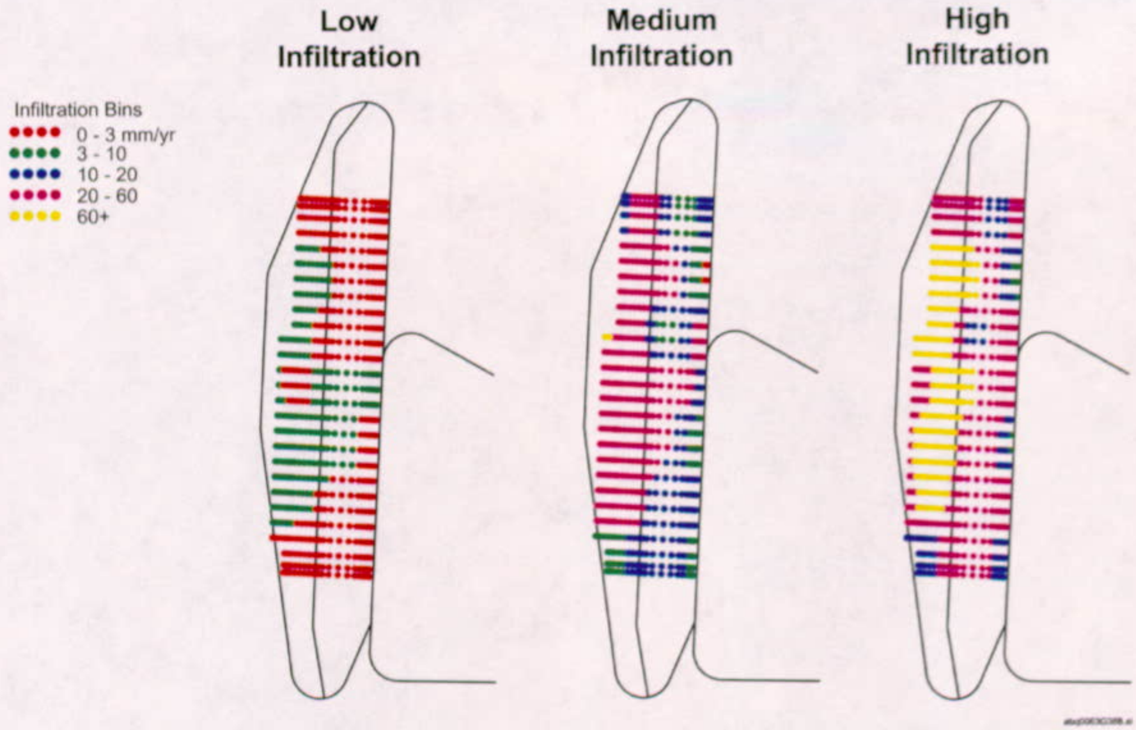
Figure 3.3-1. Information Flow Diagram for Engineered Barrier System Environments



abq0063G414.pdf

Source: Taken from CRWMS M&O 1999 [125130], Figure 1

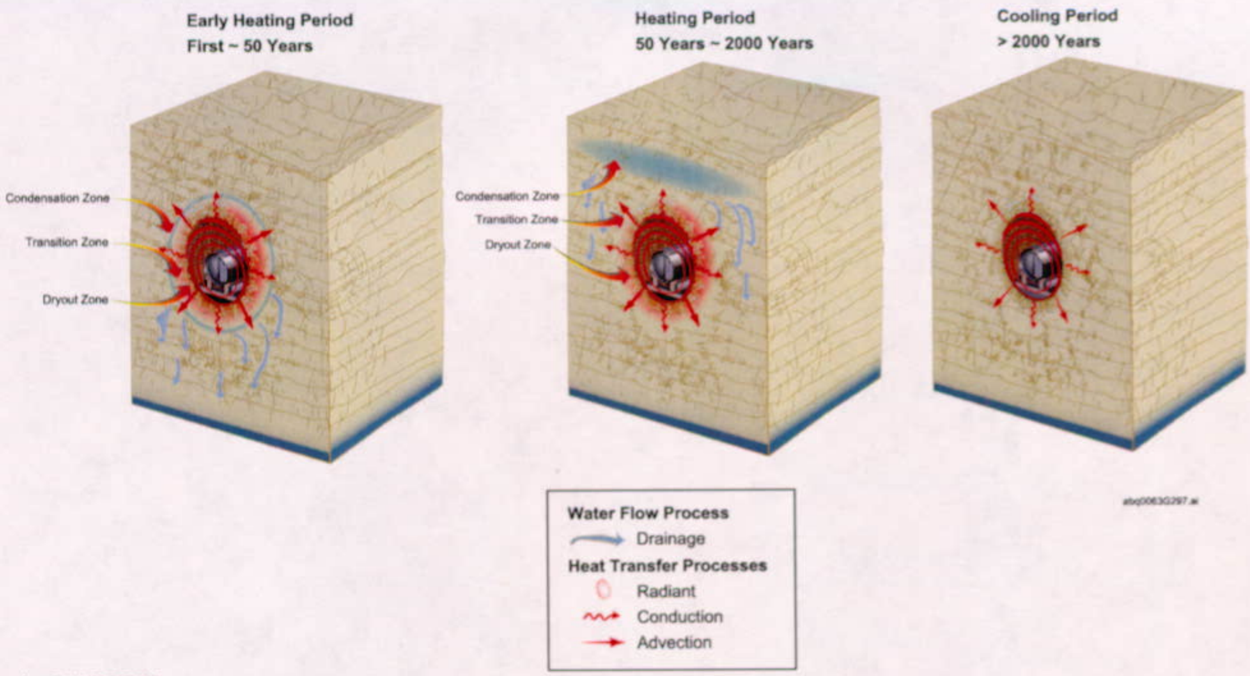
Figure 3.3-2. Illustration of a Typical Key Block and Associated Fracture Planes



abq0063G388

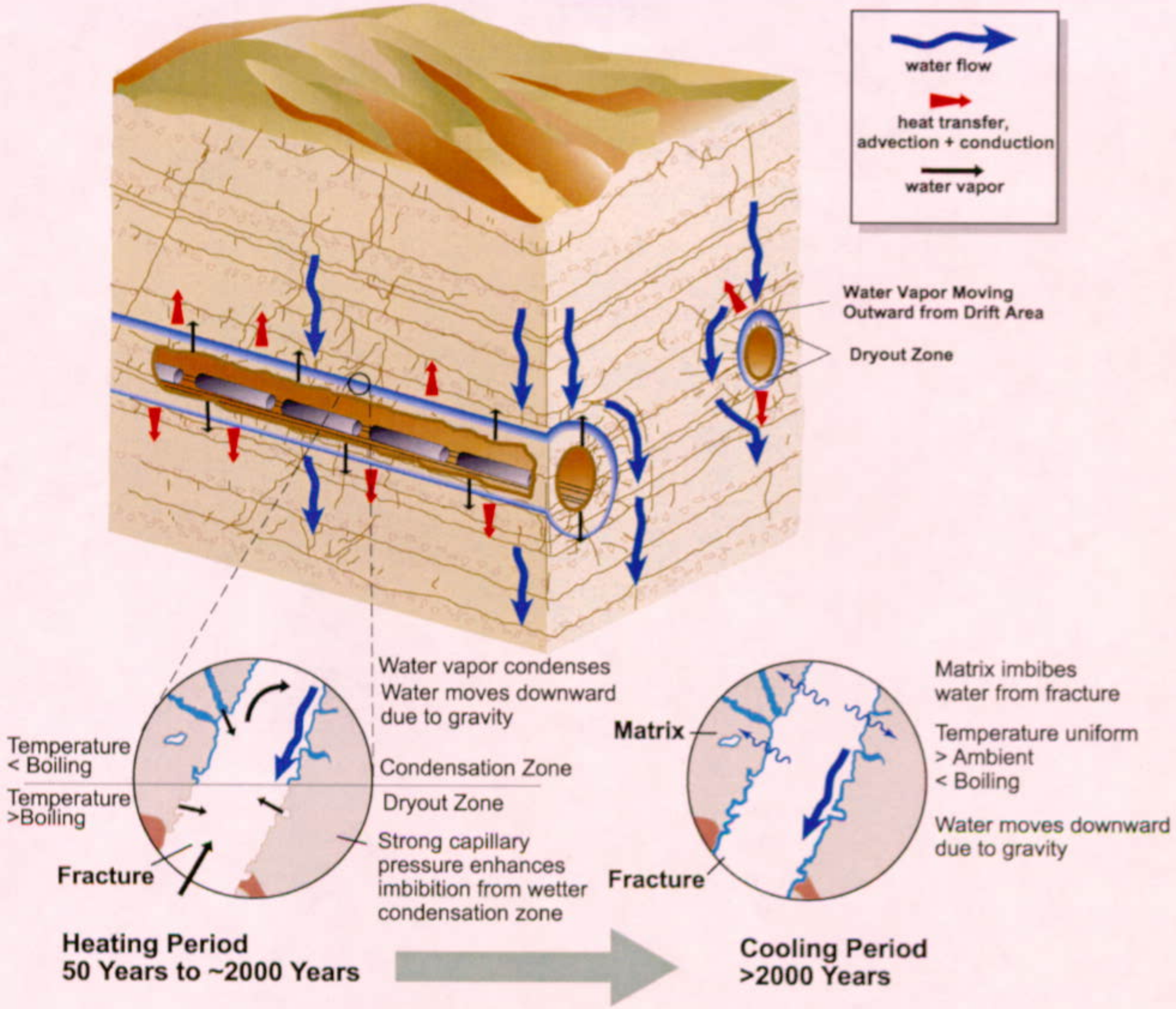
Source: Adapted from CRWMS M&O 2000 [152204], Figures 18 to 20

Figure 3.3-3. Locations of the Five Infiltration Bins for Three Infiltration Cases



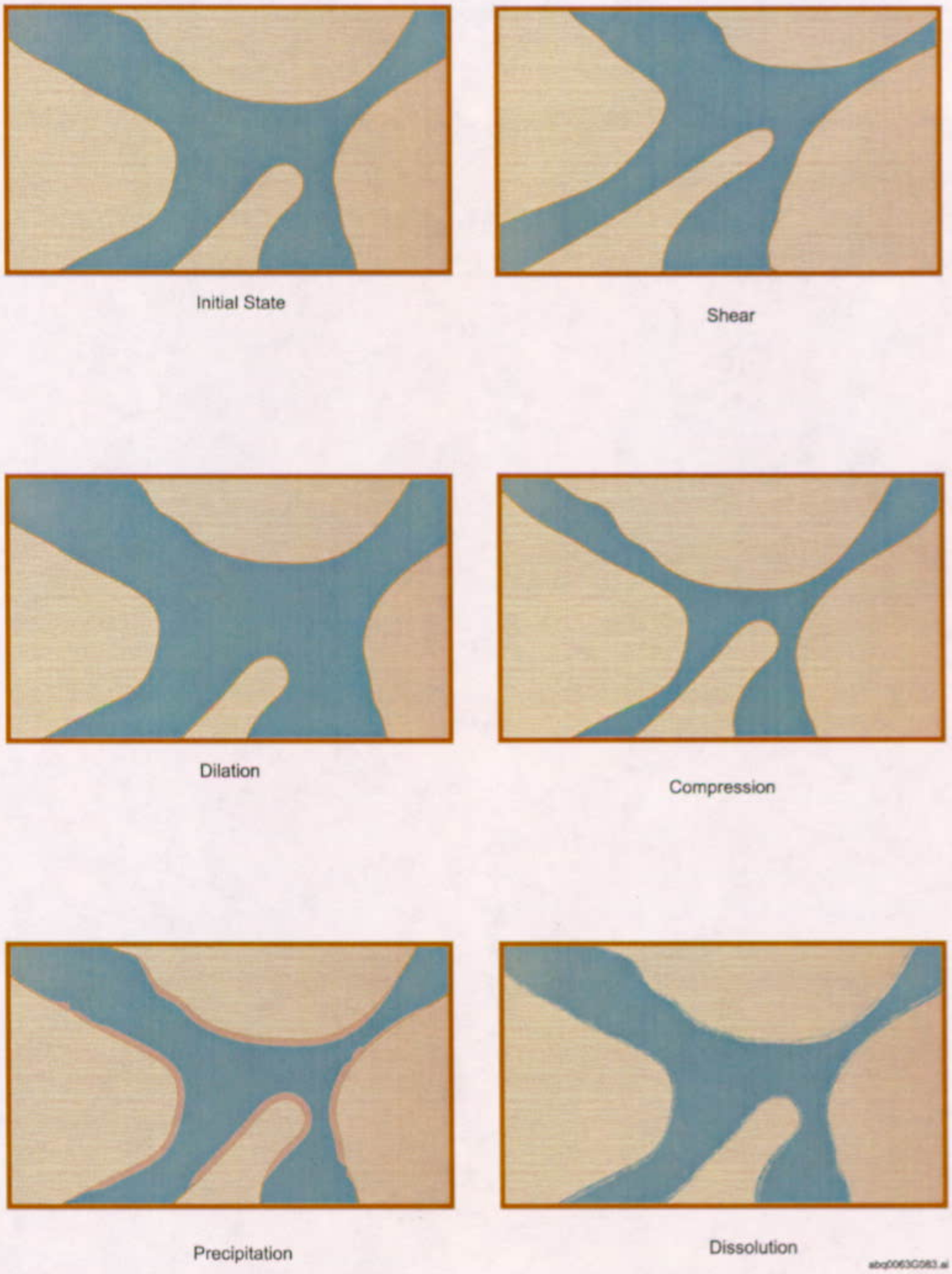
abq0063G297

Figure 3.3-4. Progression of Thermal Hydrologic Processes through Time



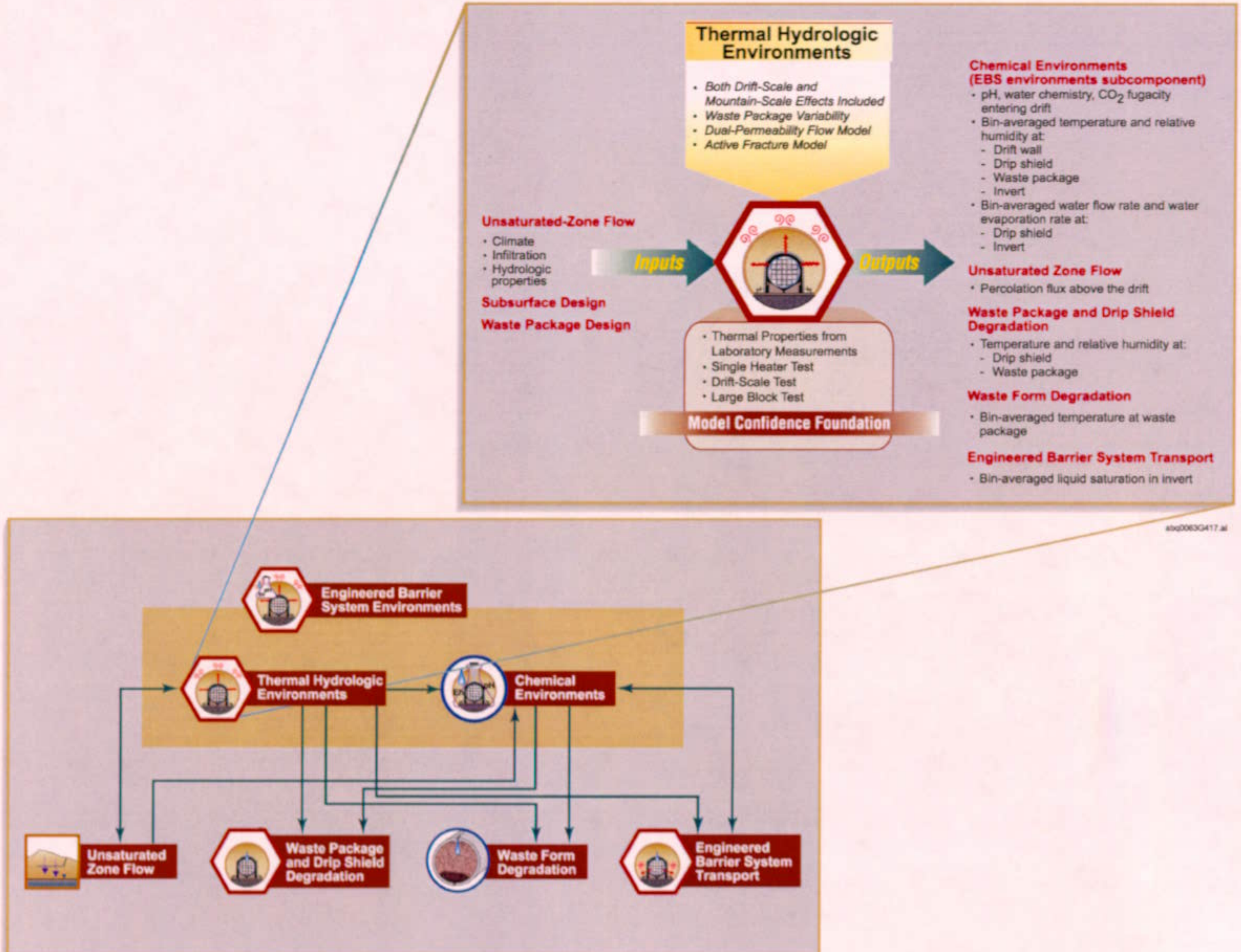
abq0063G099

Figure 3.3-5. Conceptual Drawing Illustrating Flow of Liquid Water and Water Vapor in Fractures



abq0063G083

Figure 3.3-6. Types of Coupled Process Effects on Fractures



abq0063G417

Figure 3.3-7. Connections Between Thermal Hydrologic Environments and Other Total System Performance Assessment Model Components

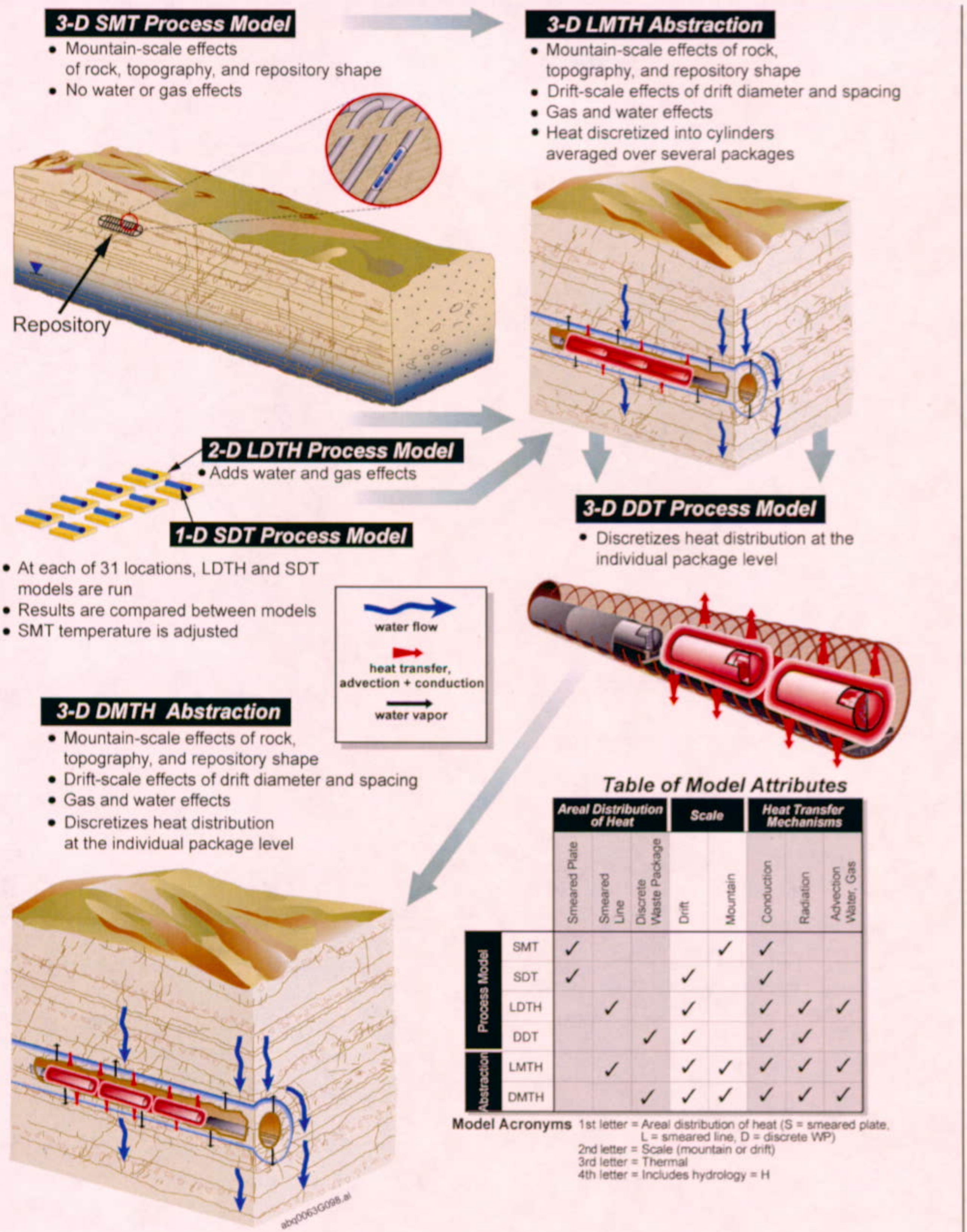
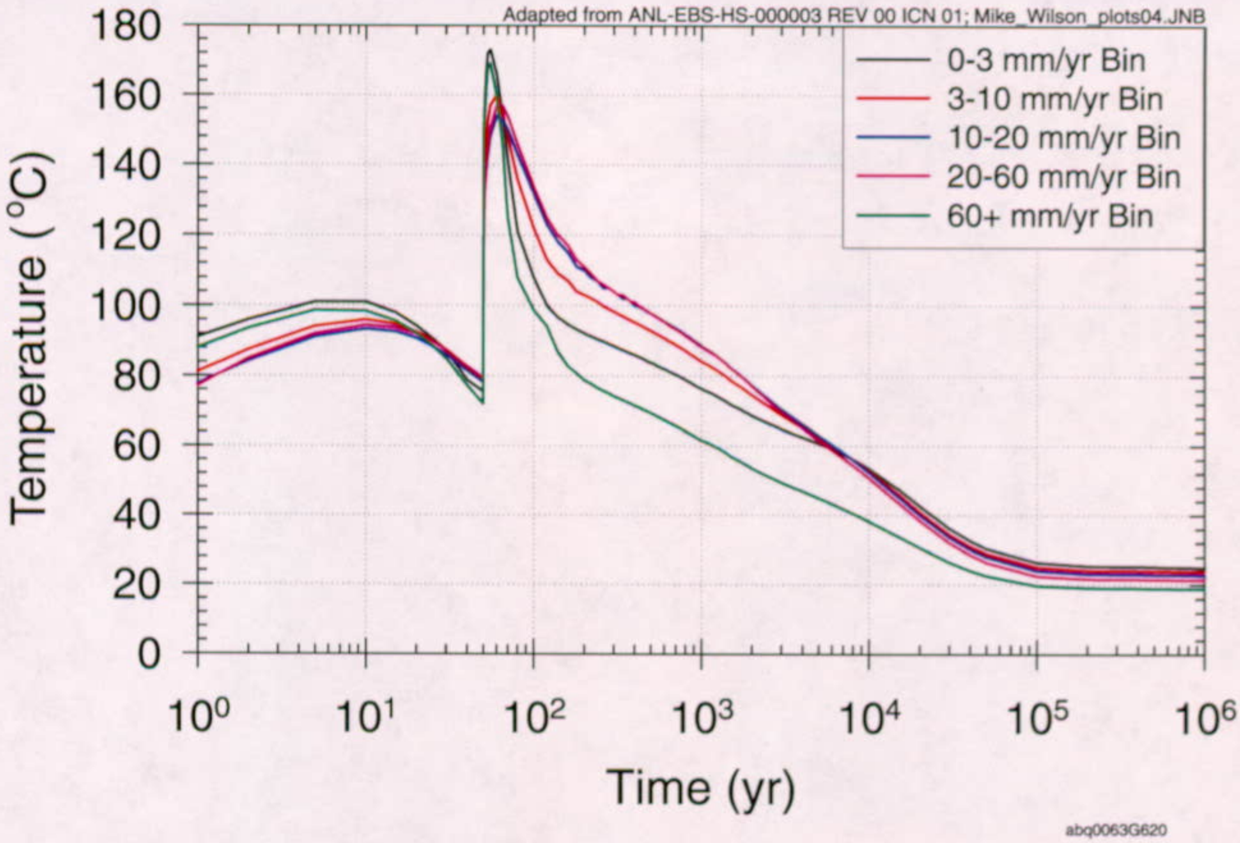


Table of Model Attributes

		Areal Distribution of Heat			Scale		Heat Transfer Mechanisms		
		Smeared Plate	Smeared Line	Discrete Waste Package	Drift	Mountain	Conduction	Radiation	Advection Water, Gas
Process Model	SMT	✓				✓	✓		
Process Model	SDT	✓			✓		✓		
Process Model	LDTH		✓		✓		✓	✓	✓
Process Model	DDT			✓	✓		✓	✓	
Abstraction	LMTH		✓		✓	✓	✓	✓	✓
Abstraction	DMTH			✓	✓	✓	✓	✓	✓

abq0063G098

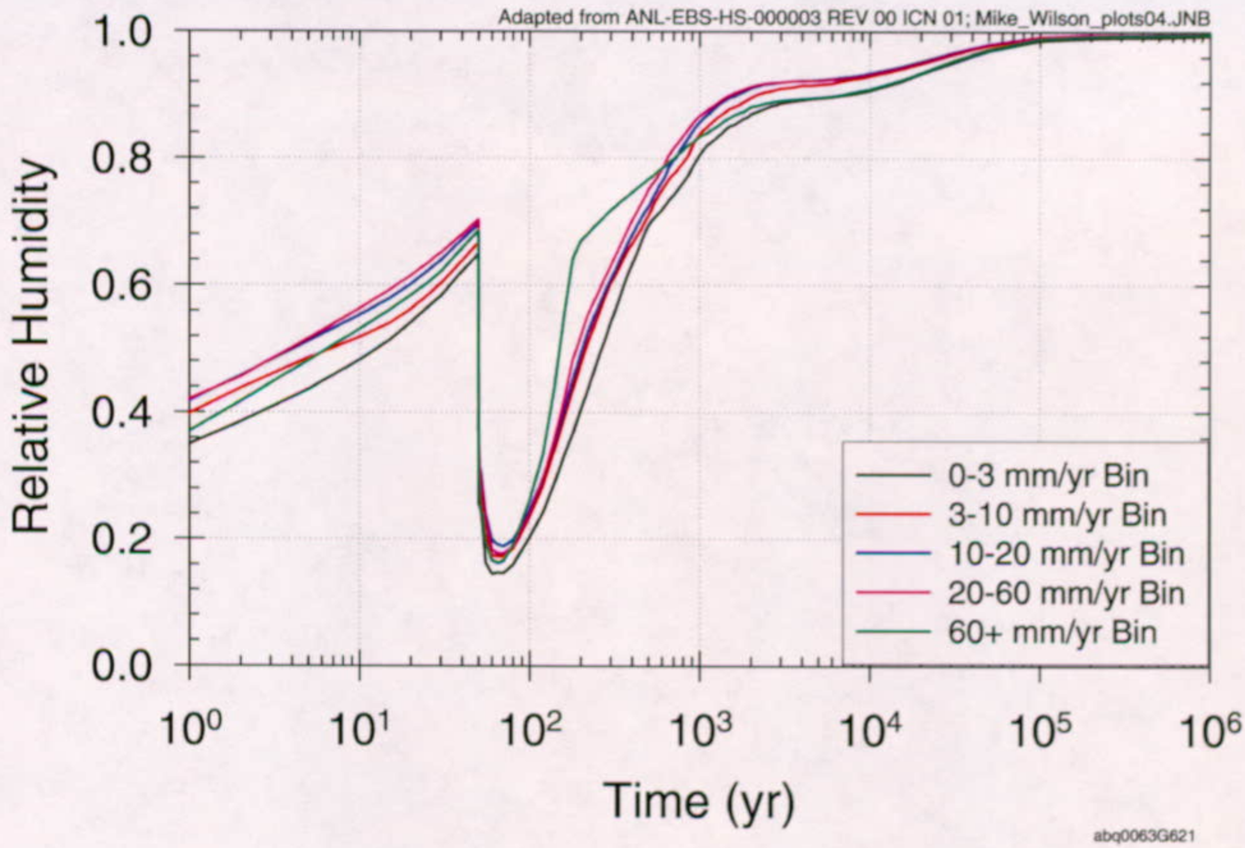
Figure 3.3-8. Illustration of the Multiscale Thermal Hydrology Model



abq0063G620

Source: Adapted from CRWMS M&O 2000 [152204], Figure 33

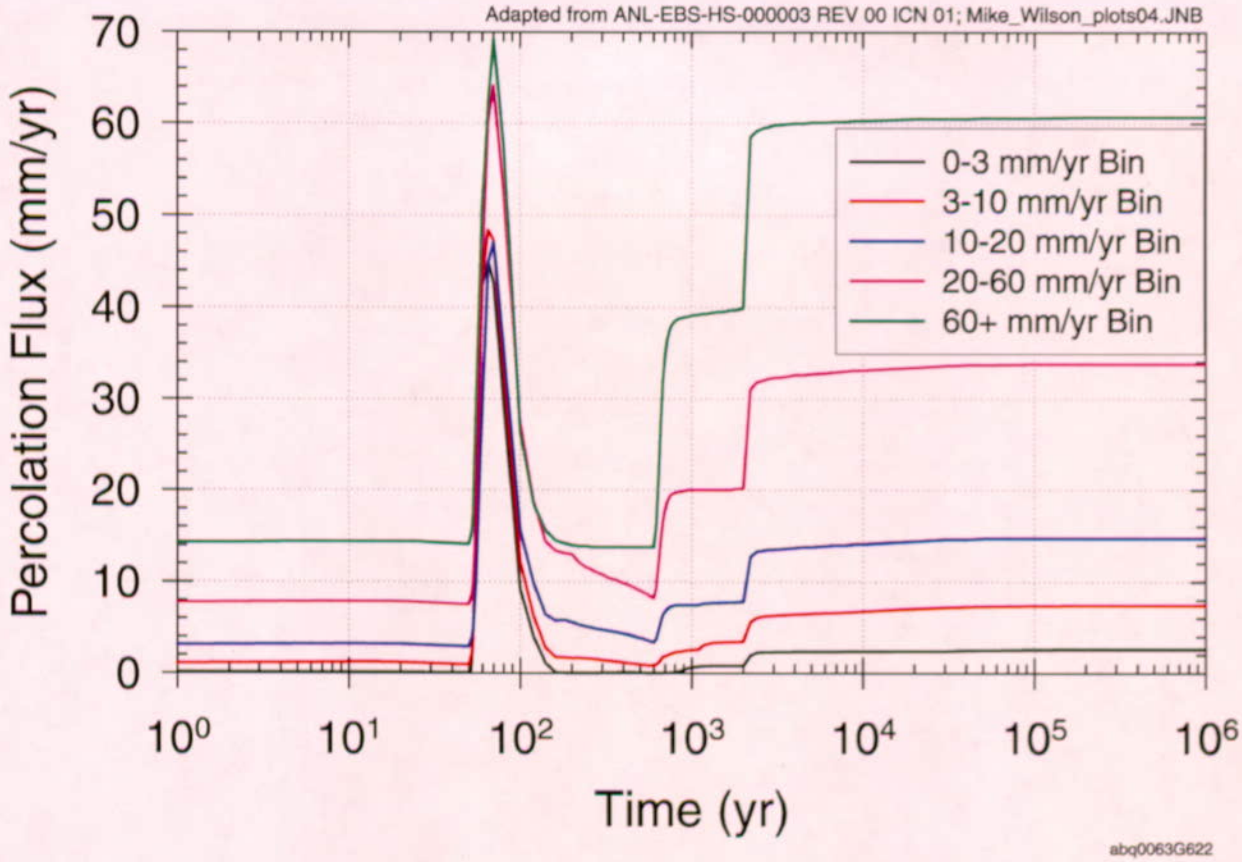
Figure 3.3-9. Bin-Averaged Waste Package Temperature, Medium-Infiltration Case



abq0063G621

Source: Adapted from CRWMS M&O 2000 [152204], Figure 36

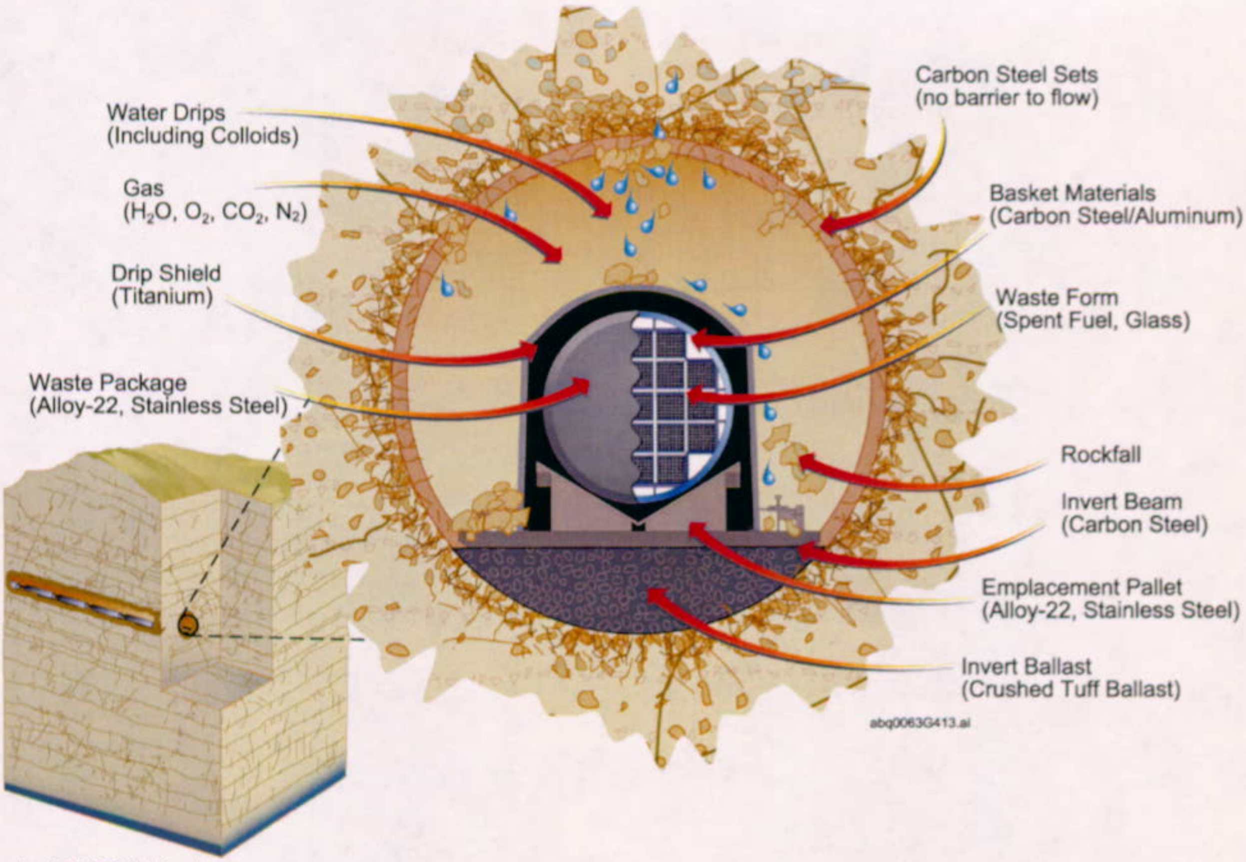
Figure 3.3-10. Bin-Averaged Waste Package Relative Humidity, Medium-Infiltration Case



abq0063G622

Source: Adapted from CRWMS M&O 2000 [149860], Figure 50

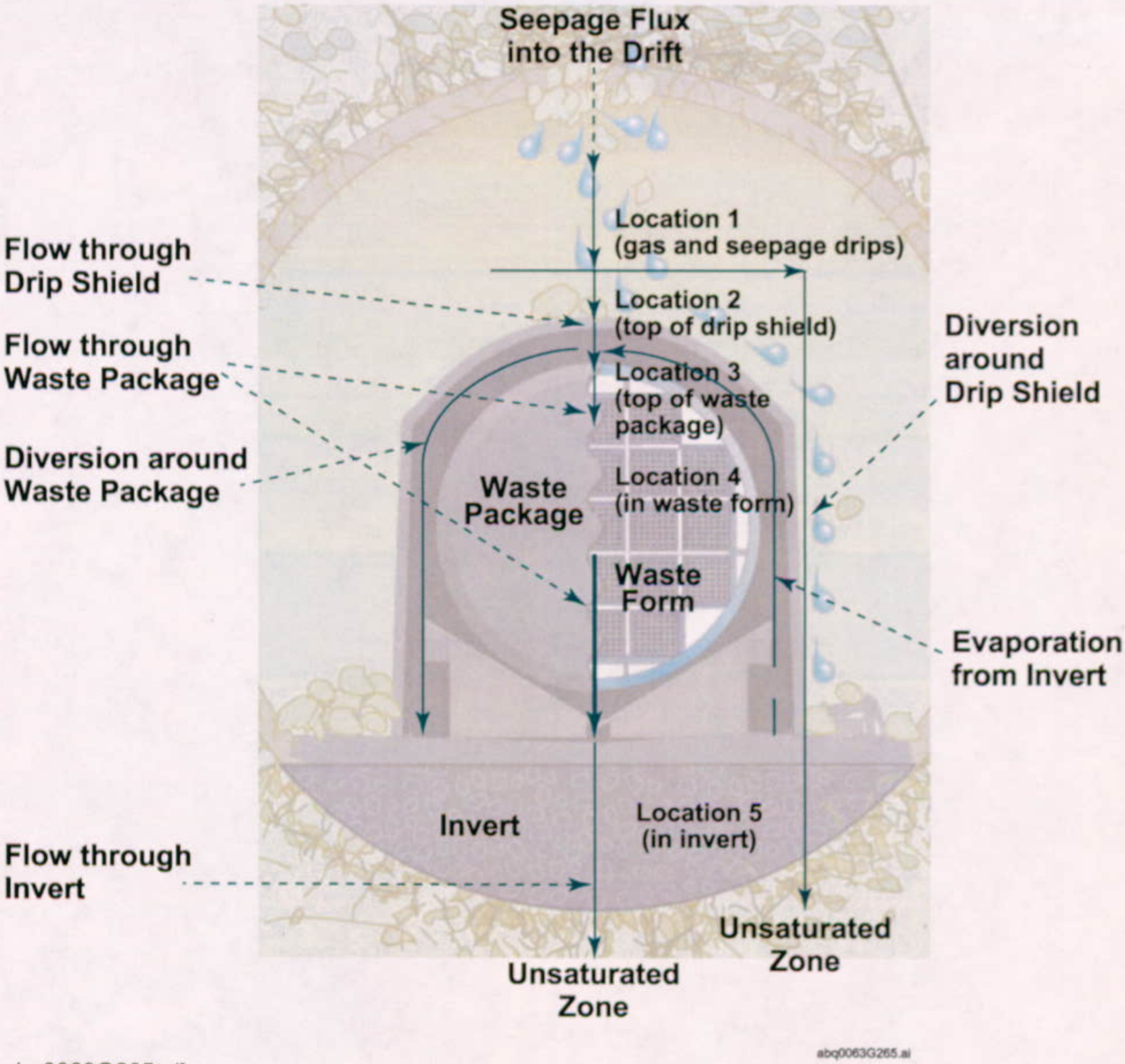
Figure 3.3-11. Bin-Averaged Percolation Flux above the Drift, Medium-Infiltration Case



abq0063G413.ai]

NOTE: This Total System Performance Assessment-Site Recommendation section describes changing compositions of fluids, colloids, and solids within the potential emplacement drifts.

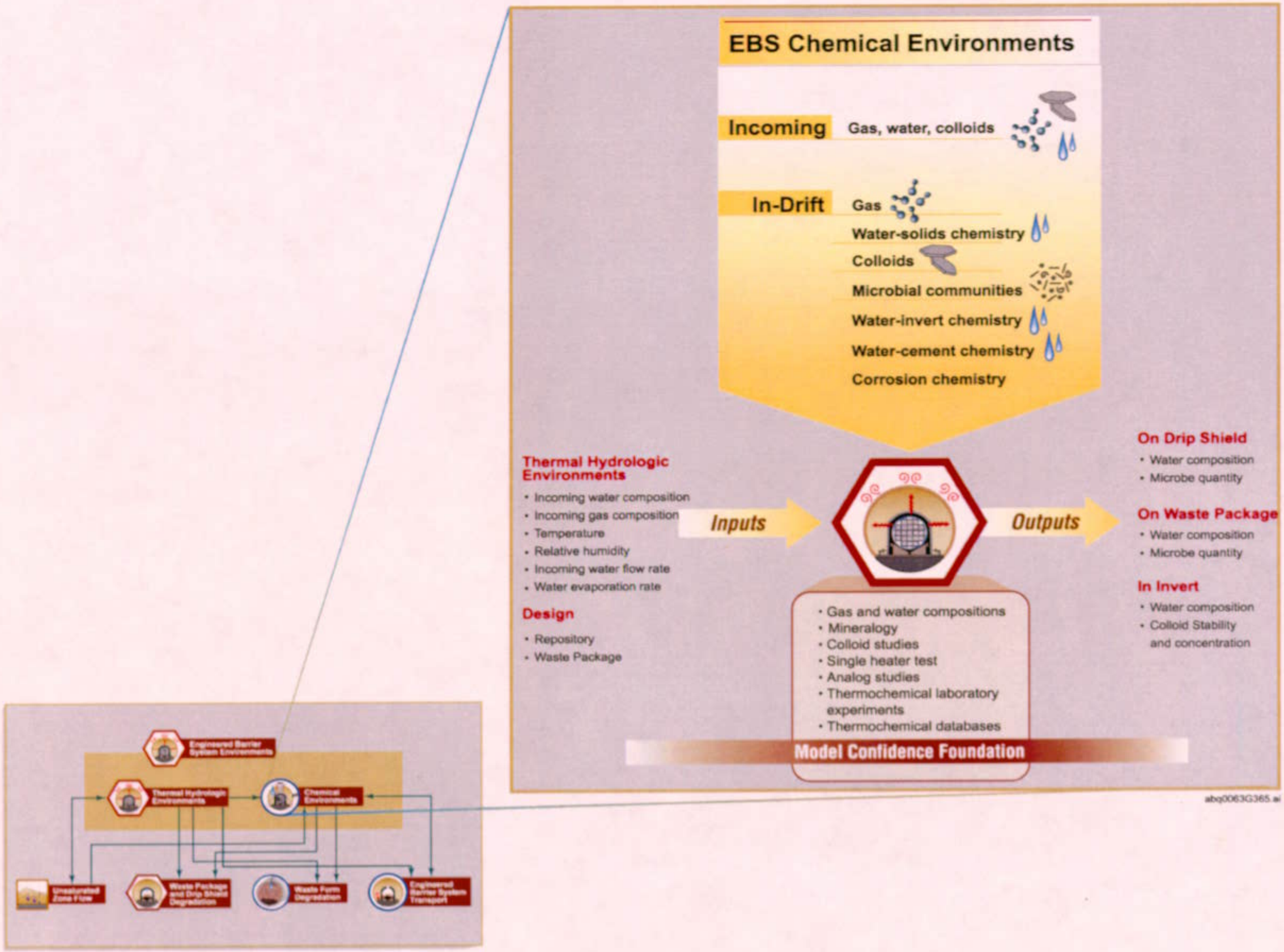
Figure 3.3-12. General Engineered Barrier System Design Features, Initial Water Movement, and Rockfall



abq0063G265.ai]

Source: Adapted from Section 3.6.

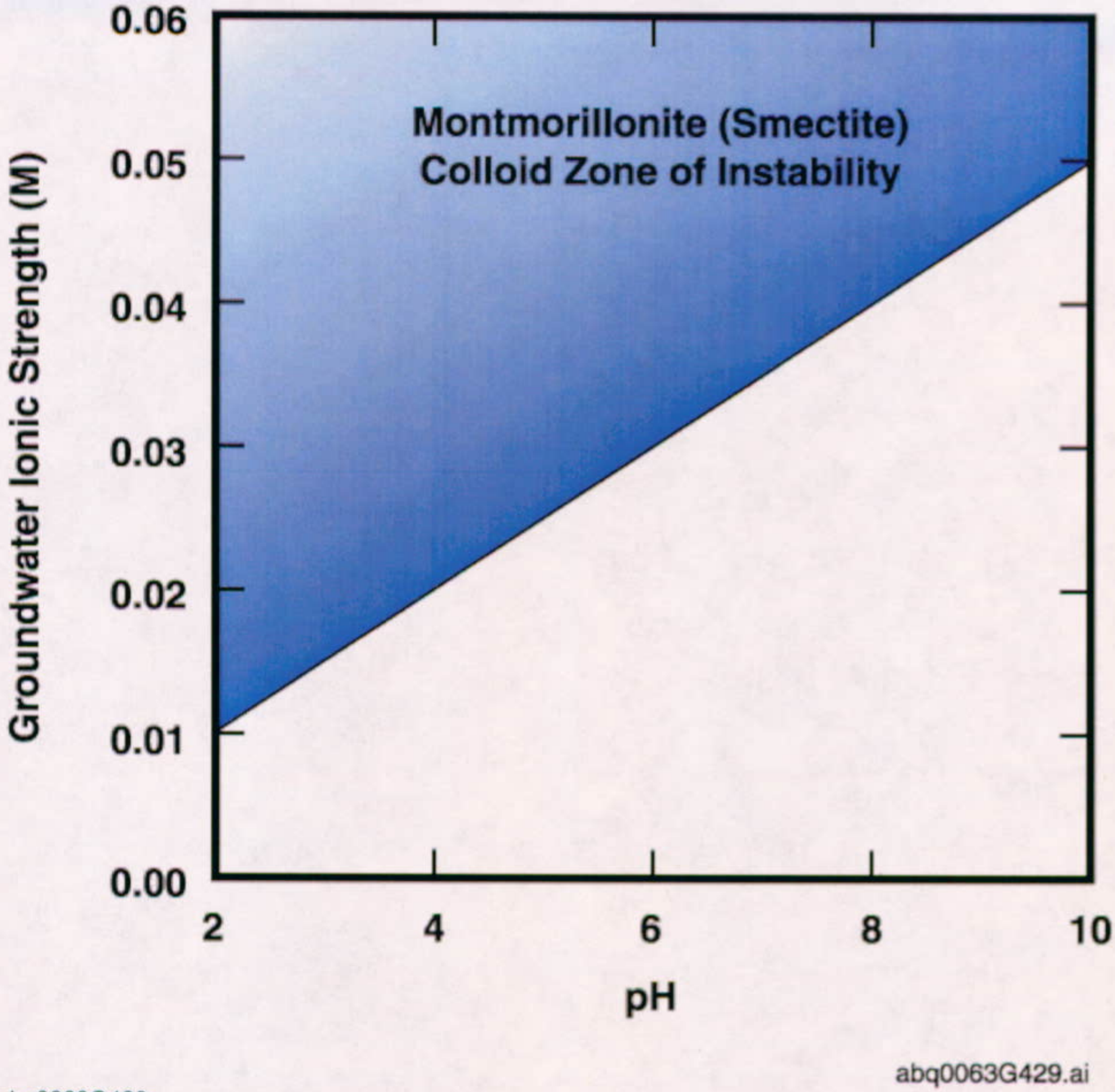
Figure 3.3-13. Schematic Diagram of Engineered Barrier System Flow Pathways (Arrows) and Critical Locations (Labels)



abq0063G365.ai

NOTE: Index figure in lower left is same as Figure 3.3-1

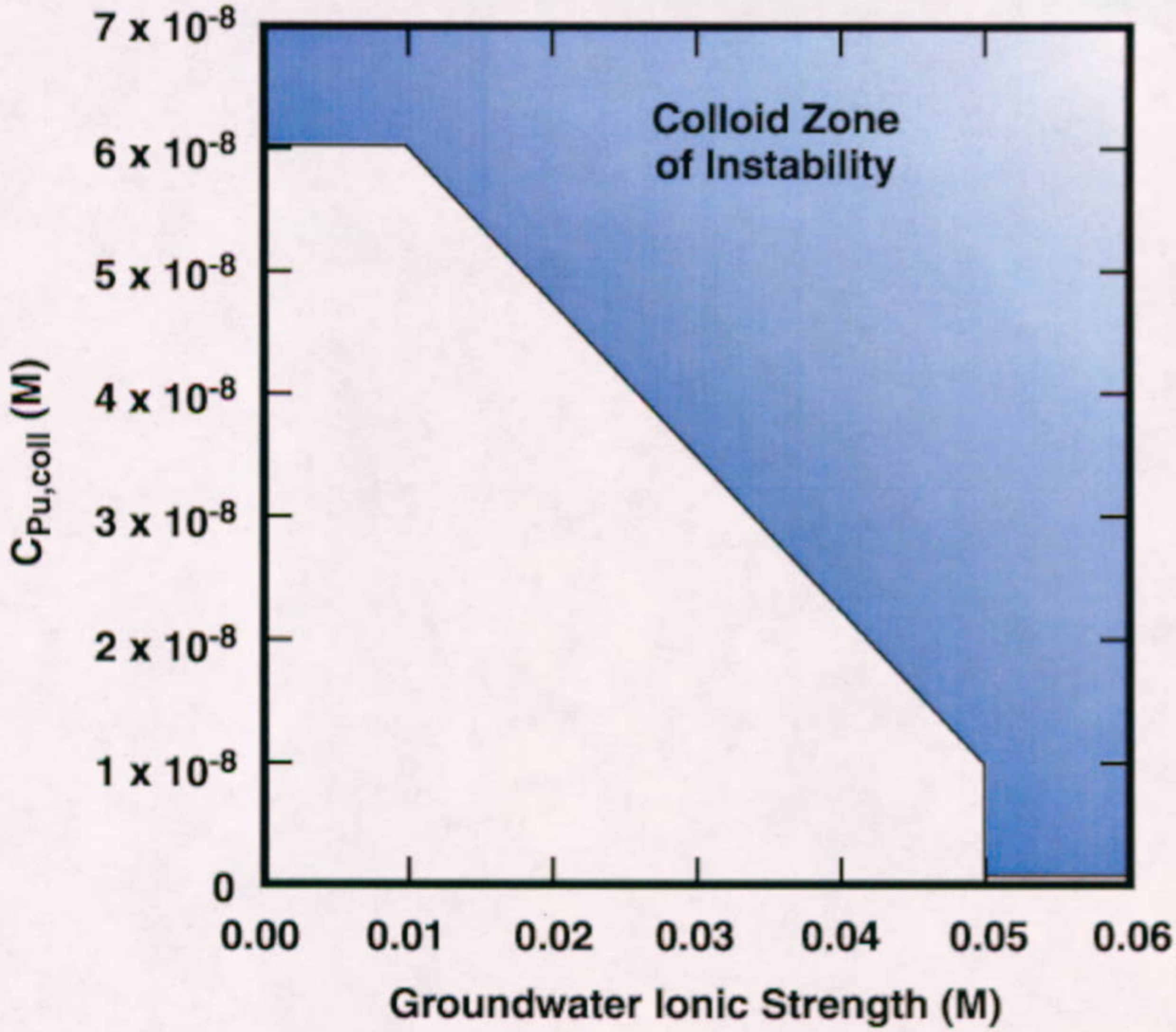
Figure 3.3-14. Engineered Barrier System Chemical Environments Model with Inputs from Thermal Hydrologic Environments and Outputs for Application at and in Engineered Barrier System Components



abq0063G429

Source: Taken from CRWMS M&O 2000 [129280], Section 6.3.4.3, Figure 4

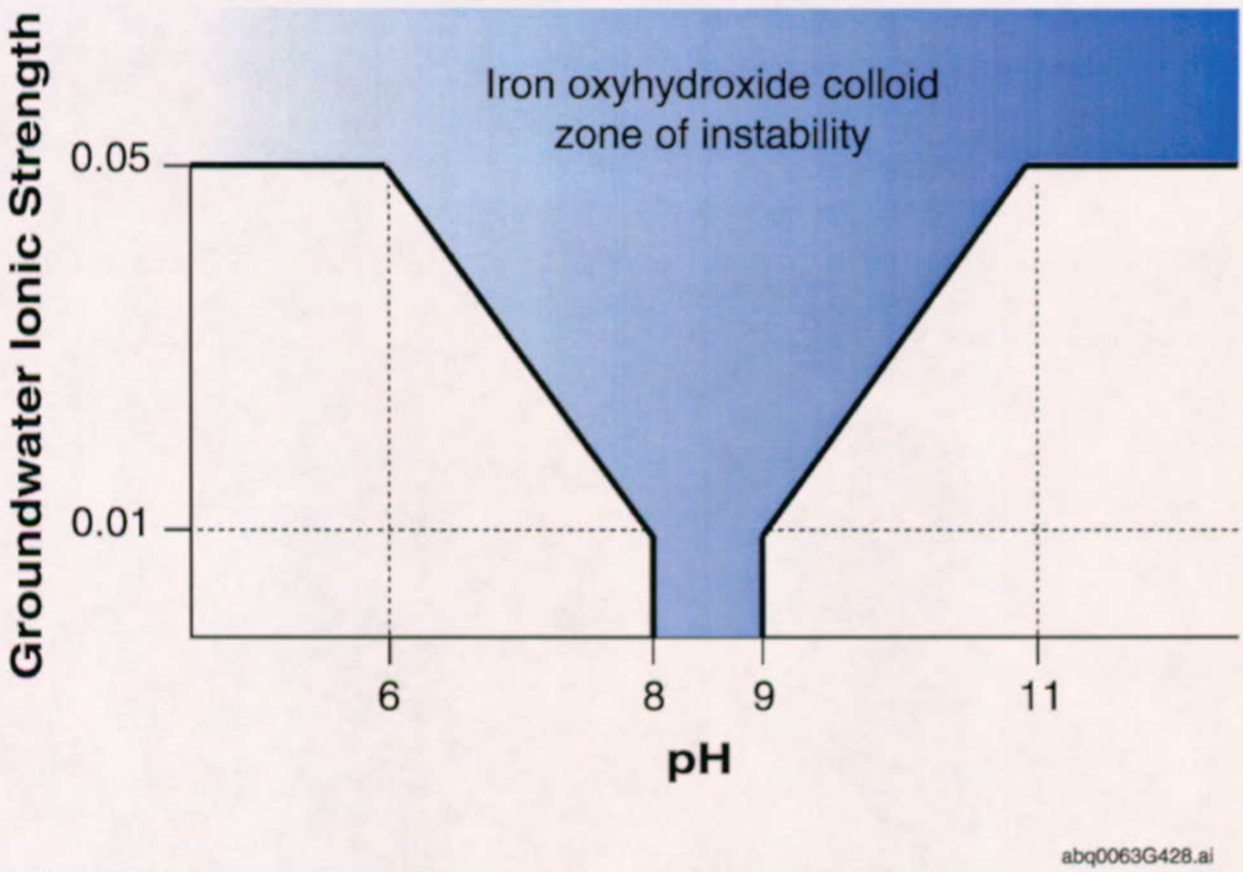
Figure 3.3-15. Schematic Representation of Smectite Stability as a Function of pH and Ionic Strength



abq0063G431

Source: Taken from CRWMS M&O 2000 [129280], Section 6.3.4.3, Figure 5

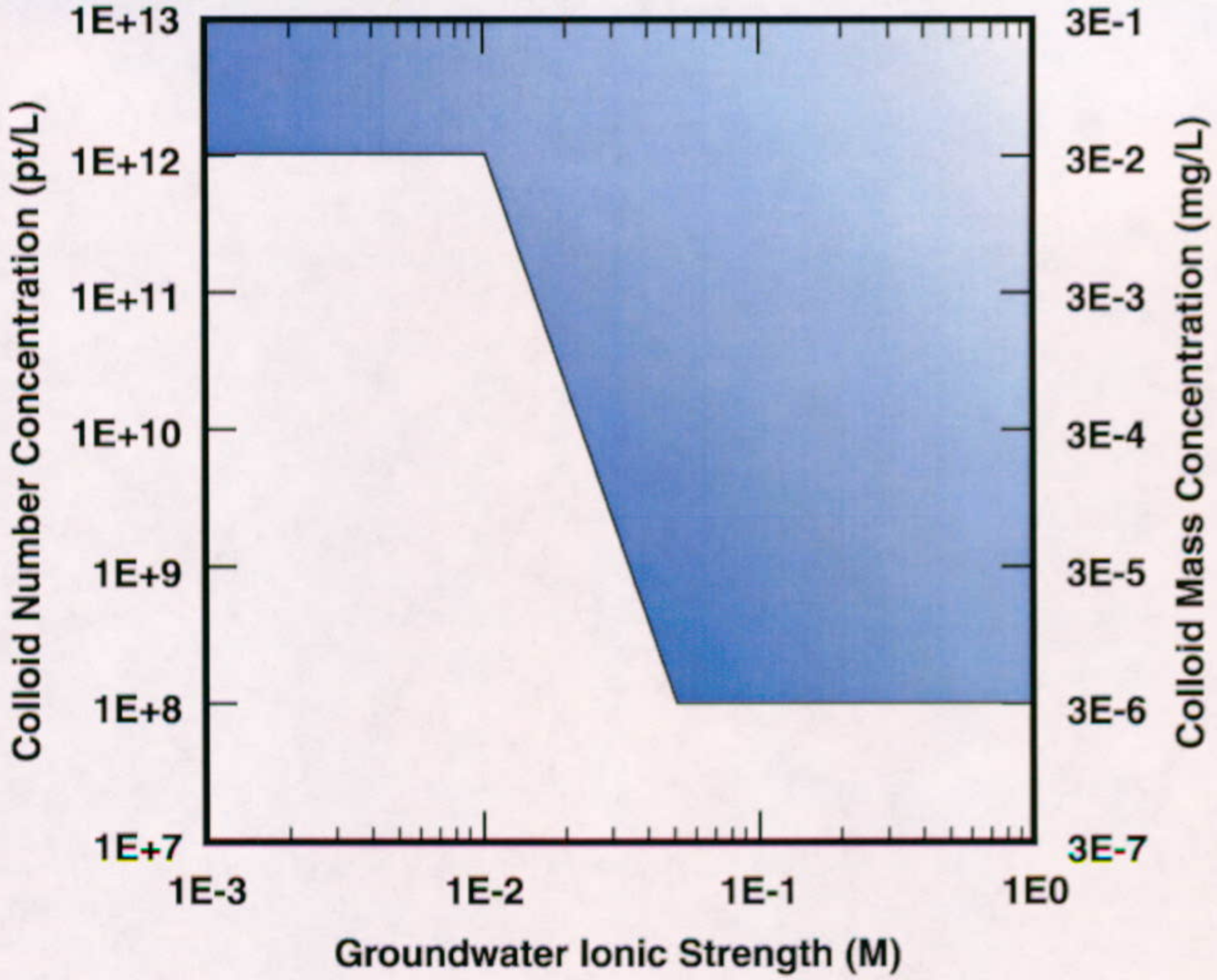
Figure 3.3-16. Schematic Relationship between Radionuclide-Bearing Colloid Concentration and Ionic Strength



abq0063G428

Source: Taken from CRWMS M&O 2000 [129280], Section 6.3.4.4, Figure 7

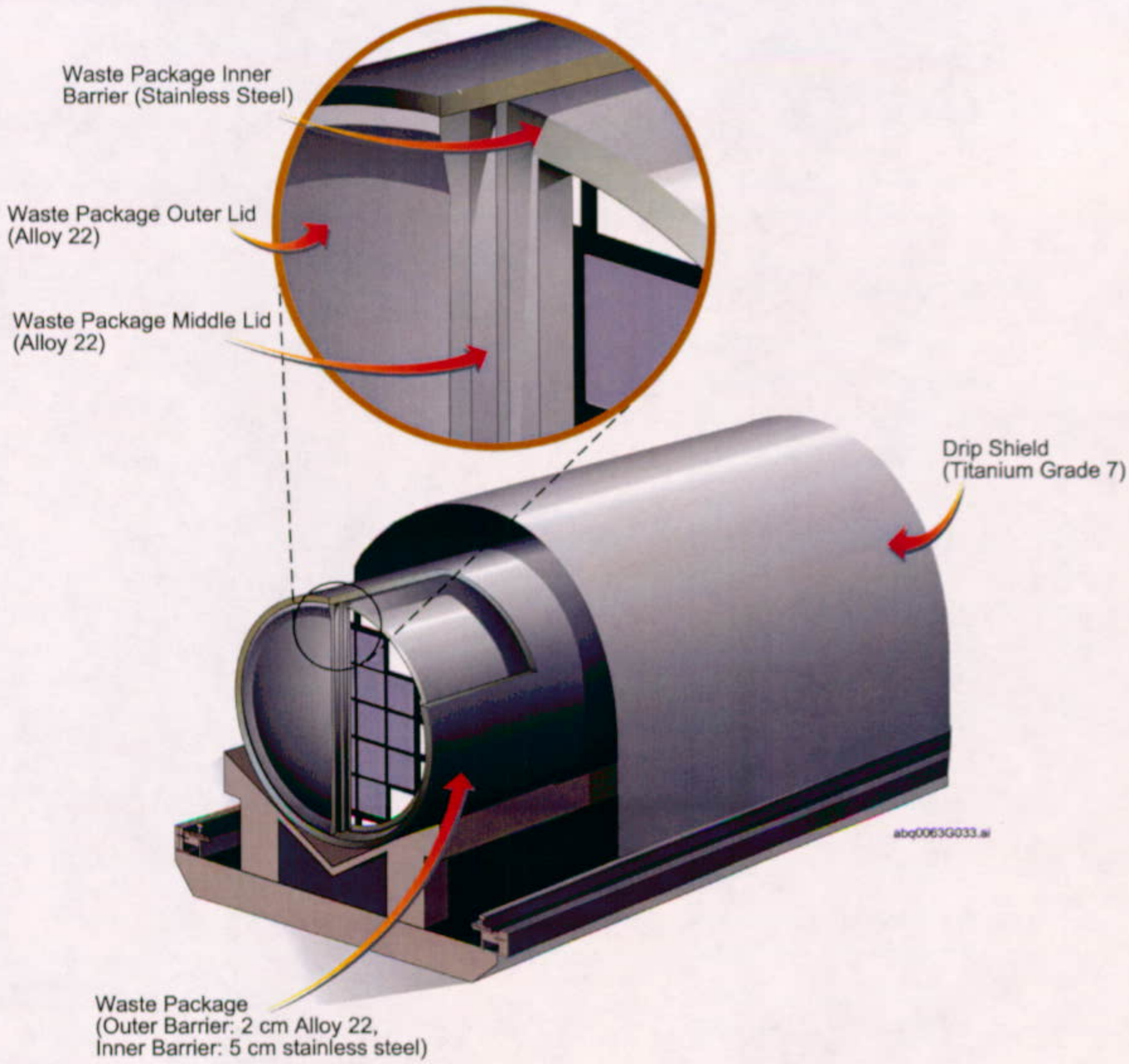
Figure 3.3-17. Schematic Representation of Iron-(Hydr)oxide Colloid Stability as a Function of pH and Ionic Strength



abq0063G430

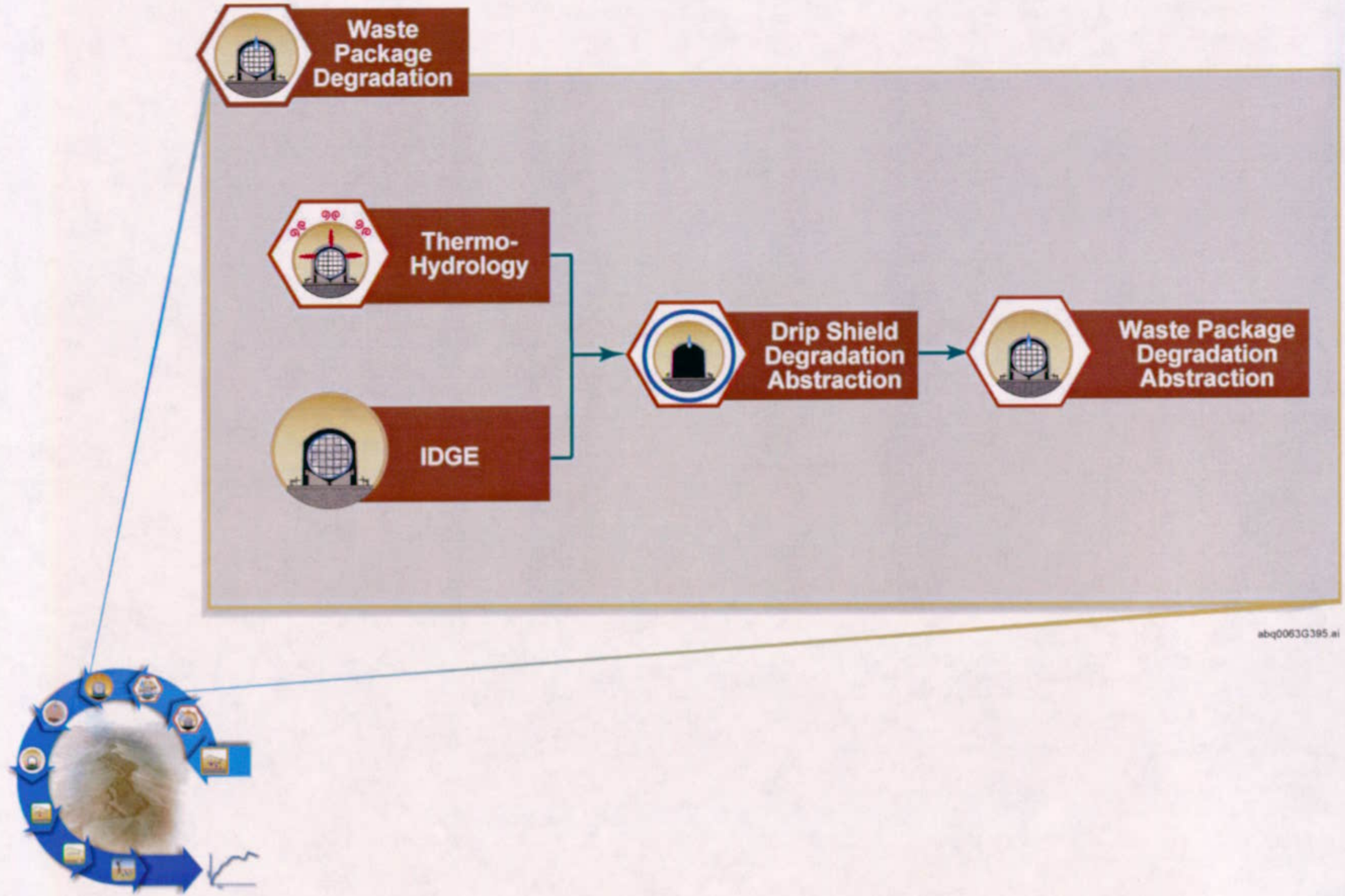
Source: Taken from CRWMS M&O 2000 [129280], Section 6.3.4.5, Figure 9

Figure 3.3-18. Schematic Relationship between Groundwater Colloid Concentration and Ionic Strength



abq0063G033

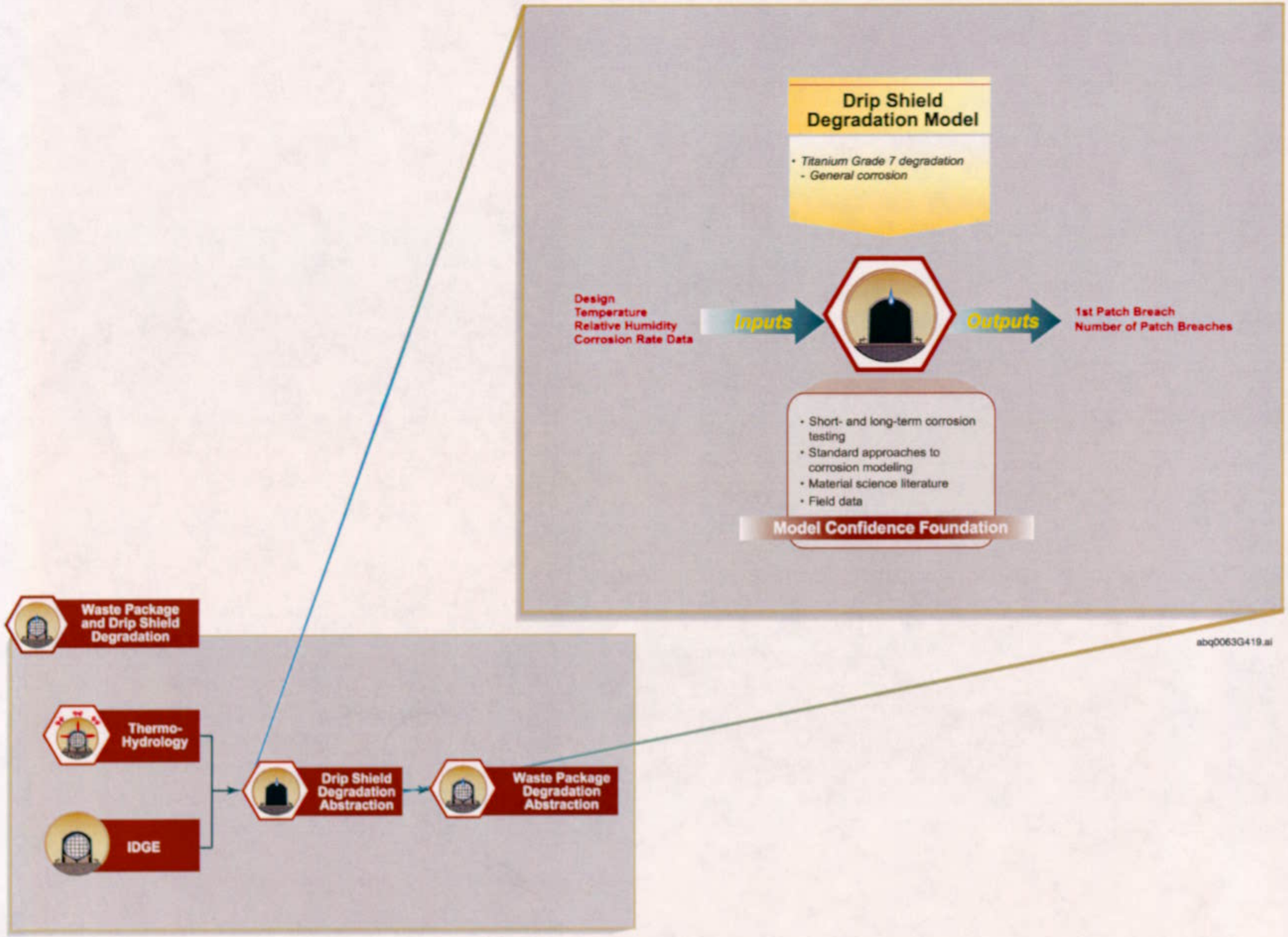
Figure 3.4-1. Schematic Design of the Drip Shield and Waste Package



abq0063G395

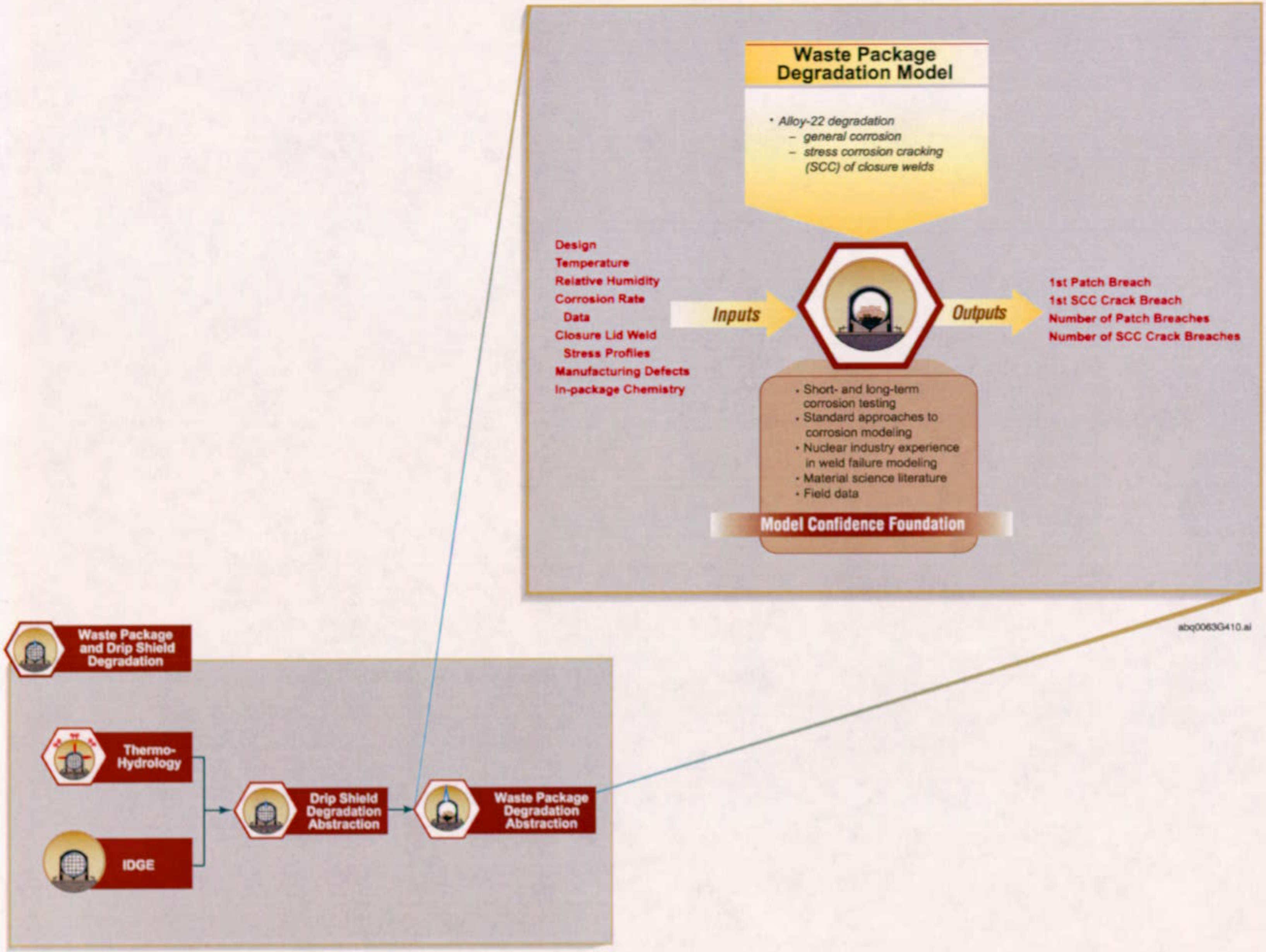
NOTE: Index figure in lower left is same as Figure 2.1-6

Figure 3.4-2. Model Data Flows for Drip Shield and Waste Package Degradation Abstraction Models



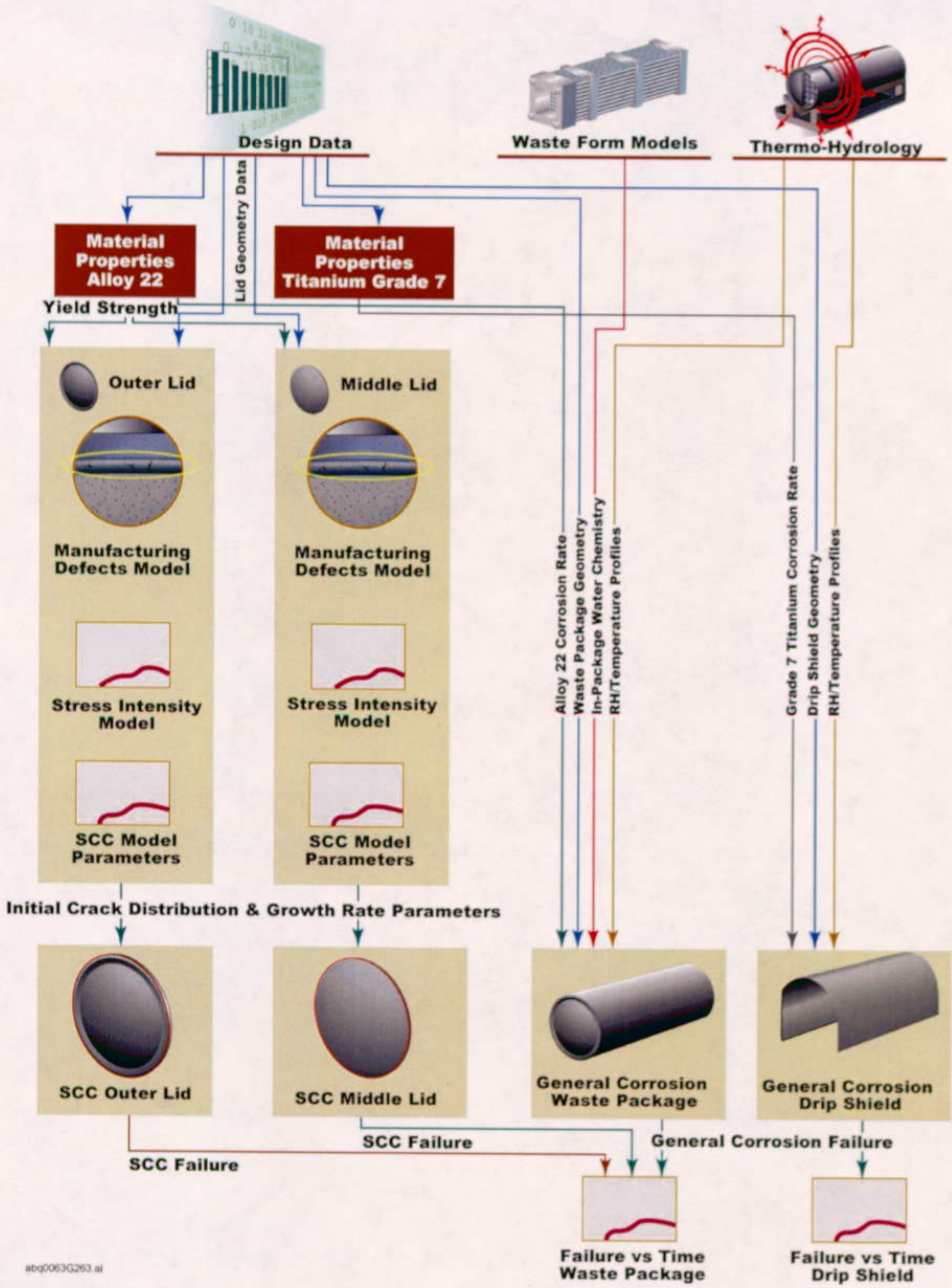
abq0063G419

Figure 3.4-3. Detail of Data Flow for Drip Shield Degradation Abstraction Model



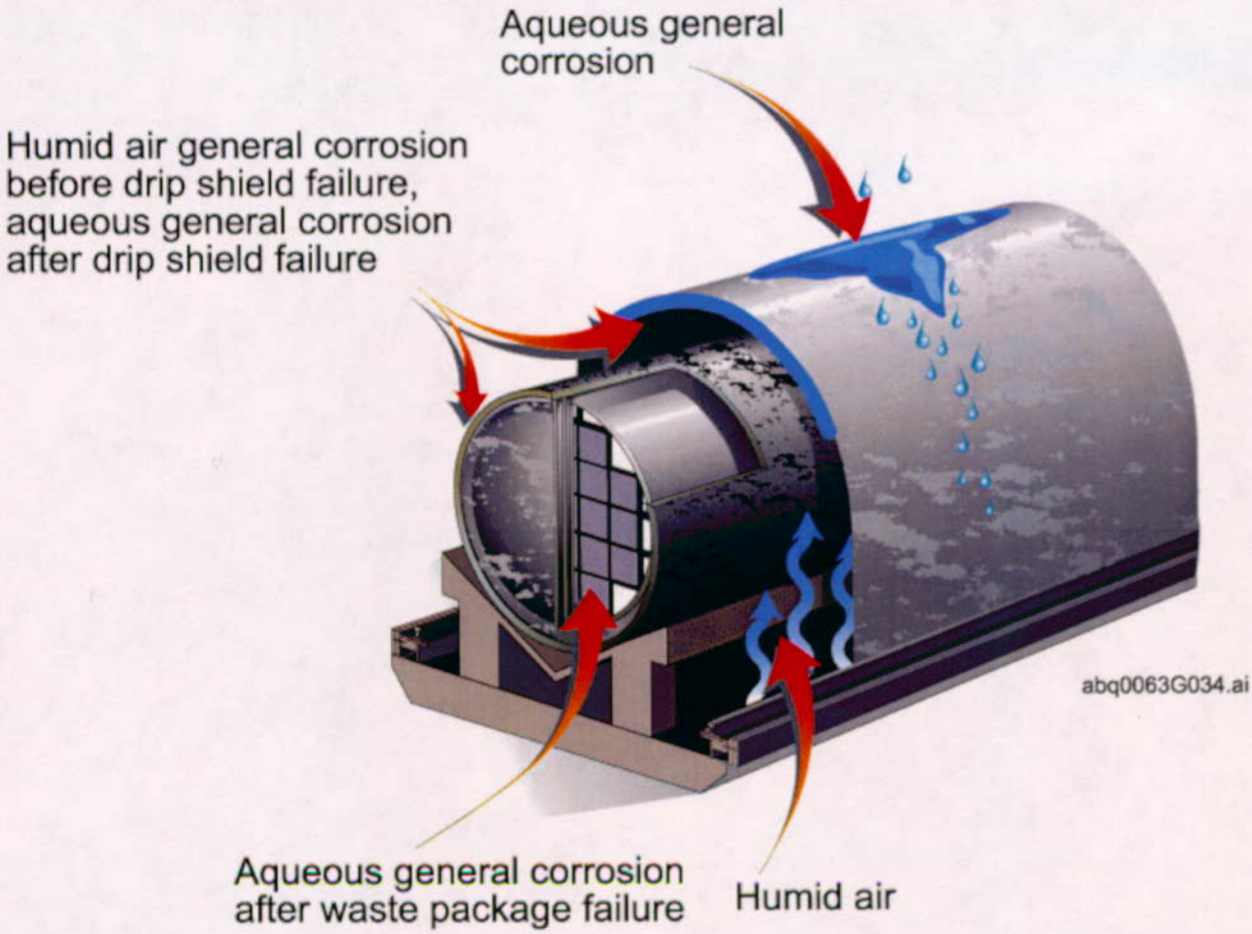
abq0063G410

Figure 3.4-4. Detail of Data Flow for Waste Package Degradation Abstraction Model



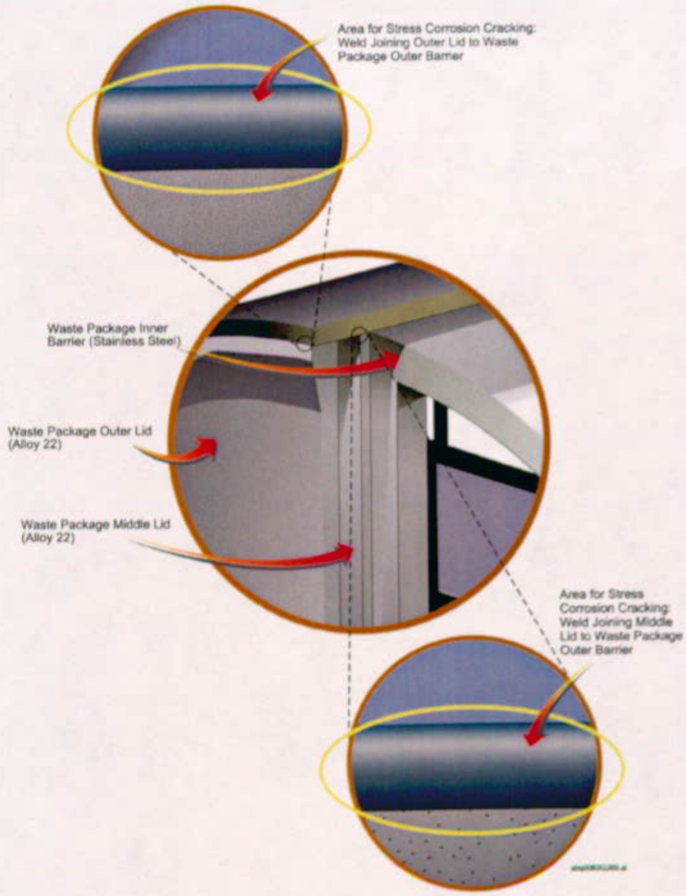
abq0063G263

Figure 3.4-5. Process and Data Flows for Drip Shield and Waste Package Degradation Conceptual Model



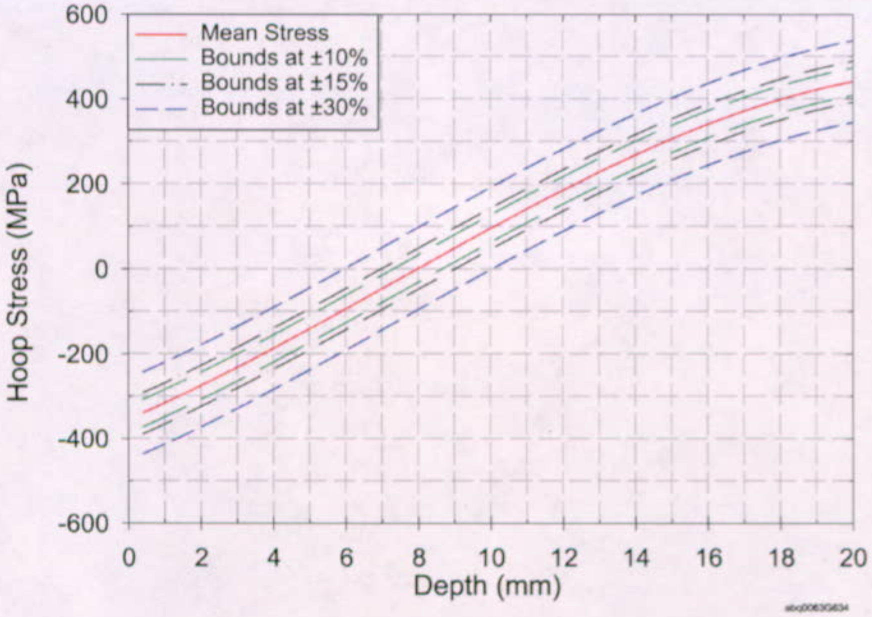
abq0063G034

Figure 3.4-6. General Corrosion Processes for Drip Shield and Waste Package



abq0063G269

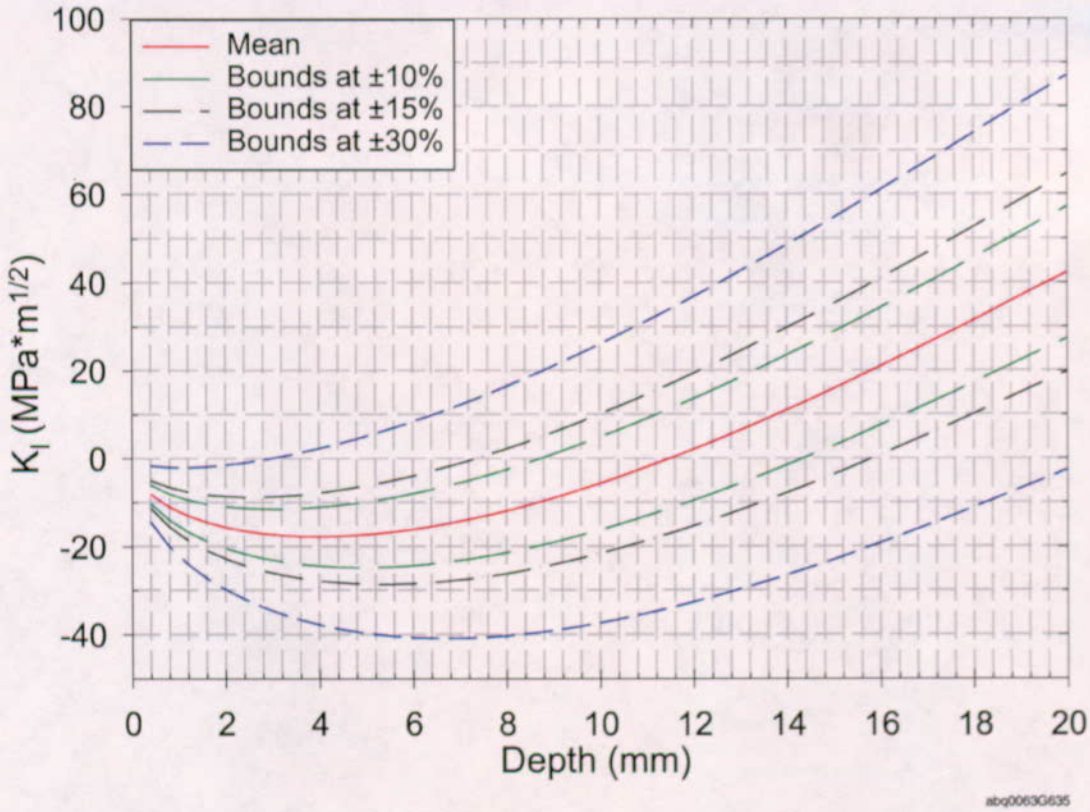
Figure 3.4-7. Schematic of Dual Closure Lid Design



DTN: MO0010MWDSUP04.010 [152884]

abq0063G634

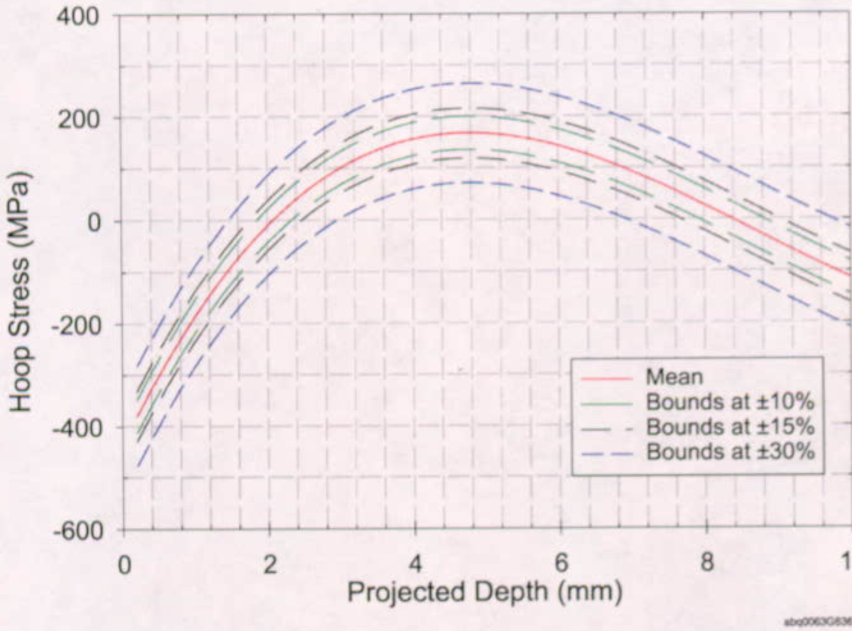
Figure 3.4-8. Hoop Stress vs. Depth for Middle Lid at 0° Angle



DTN: MO0010MWDSUP04.010 [152884]

abq0063G635

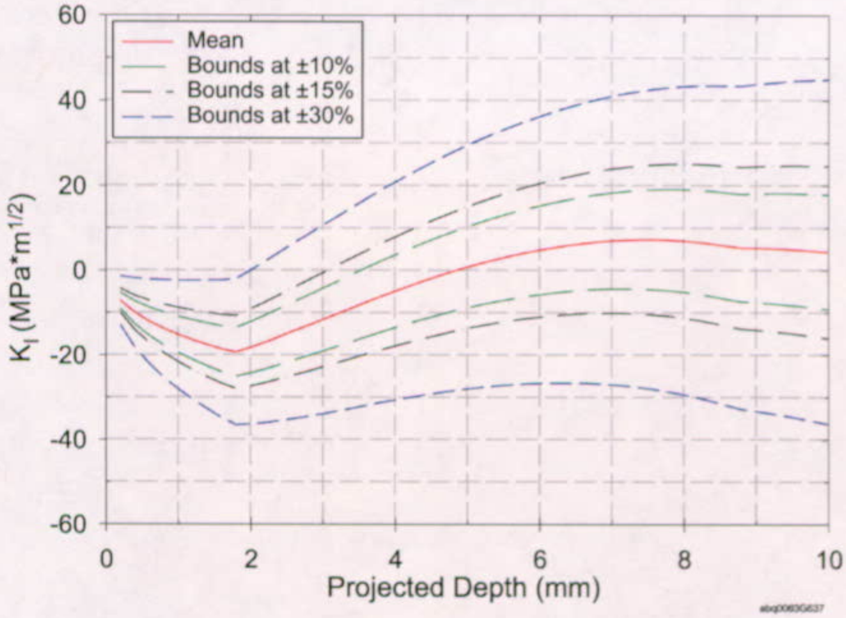
Figure 3.4-9. Stress Intensity vs. Depth for Outer Lid at 0° Angle



DTN: MO0010MWDSUP04.010 [152884]

abq0063G636

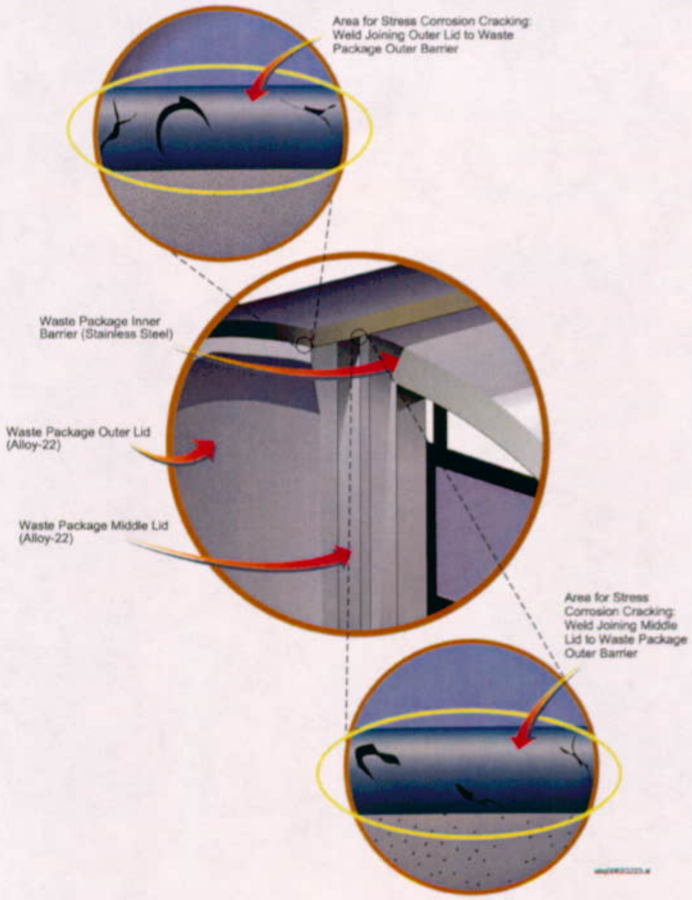
Figure 3.4-10. Hoop Stress vs. Depth for Middle Lid at 0° Angle



DTN: MO0010MWDSUP04.010 [152884]

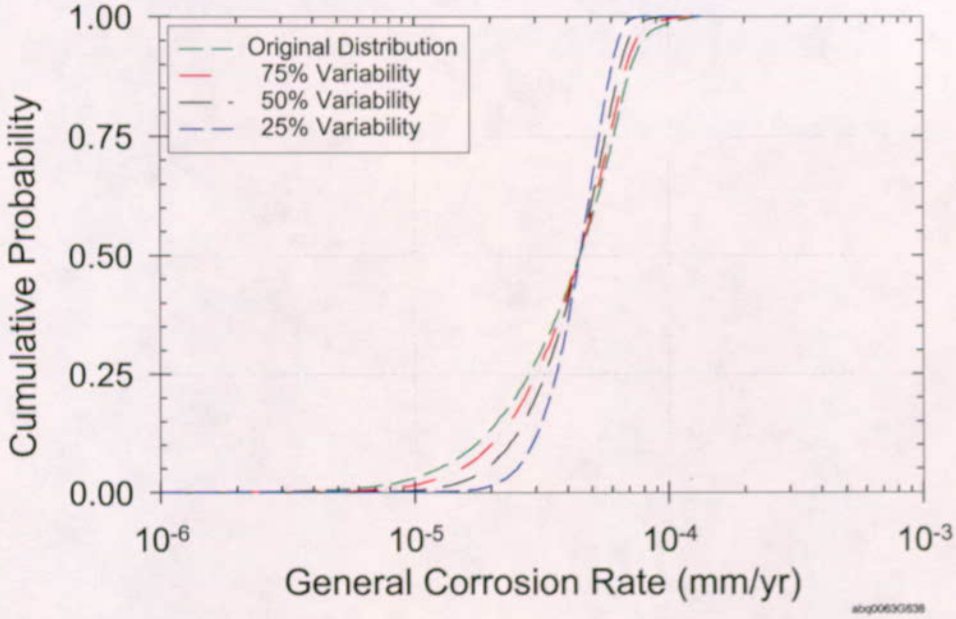
abq0063G637

Figure 3.4-11. Stress Intensity vs. Depth for Middle Lid at 0° Angle



abq0063G223

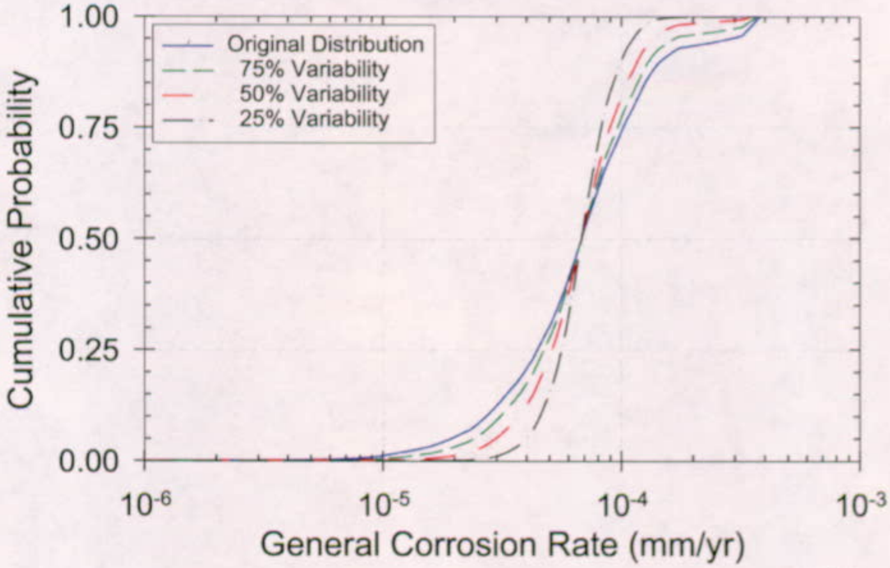
Figure 3.4-12. Closure Lid Weld Manufacturing Defect Schematic



DTN: MO0010SPASIL02.002 [152605]

abq0063G638

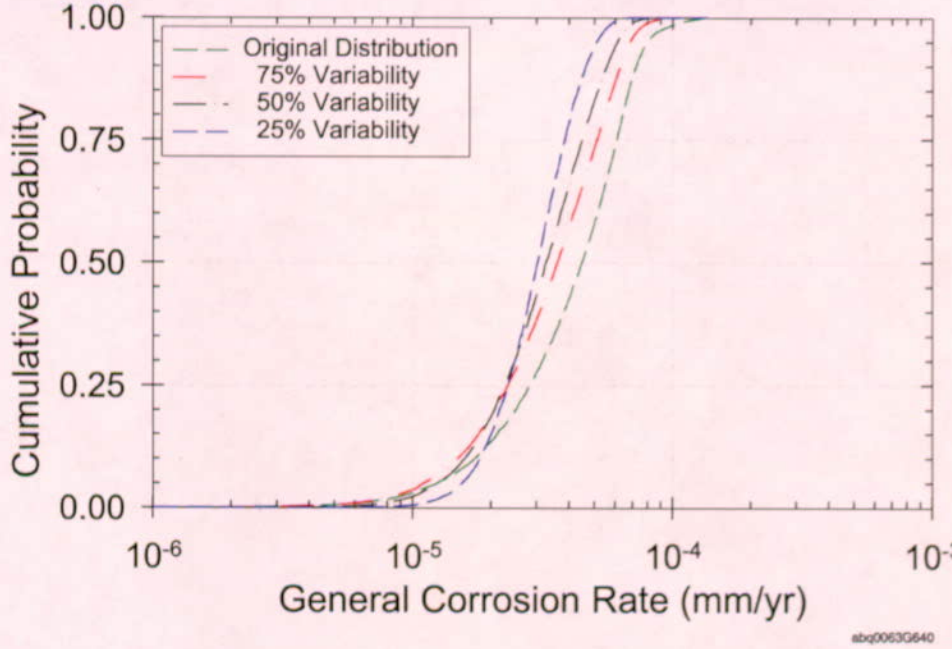
Figure 3.4-13. Variability CDFs for Alloy-22 with 75, 50, and 25 Percent Variability Using Median Uncertainty Quantile



DTN: MO0010SPASIL02.002 [152605]

abq0063G639

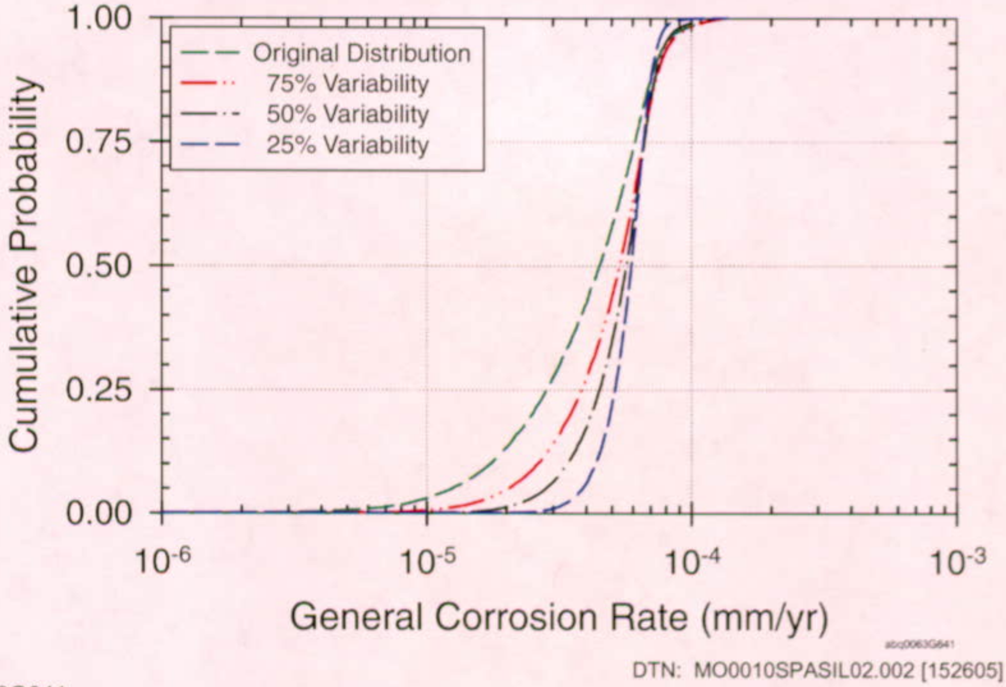
Figure 3.4-14. Variability CDFs for Titanium Grade 7 with 75, 50, and 25 Percent Variability Using Median Uncertainty Quantile



DTN: MO0010SPASIL02.002 [152605]

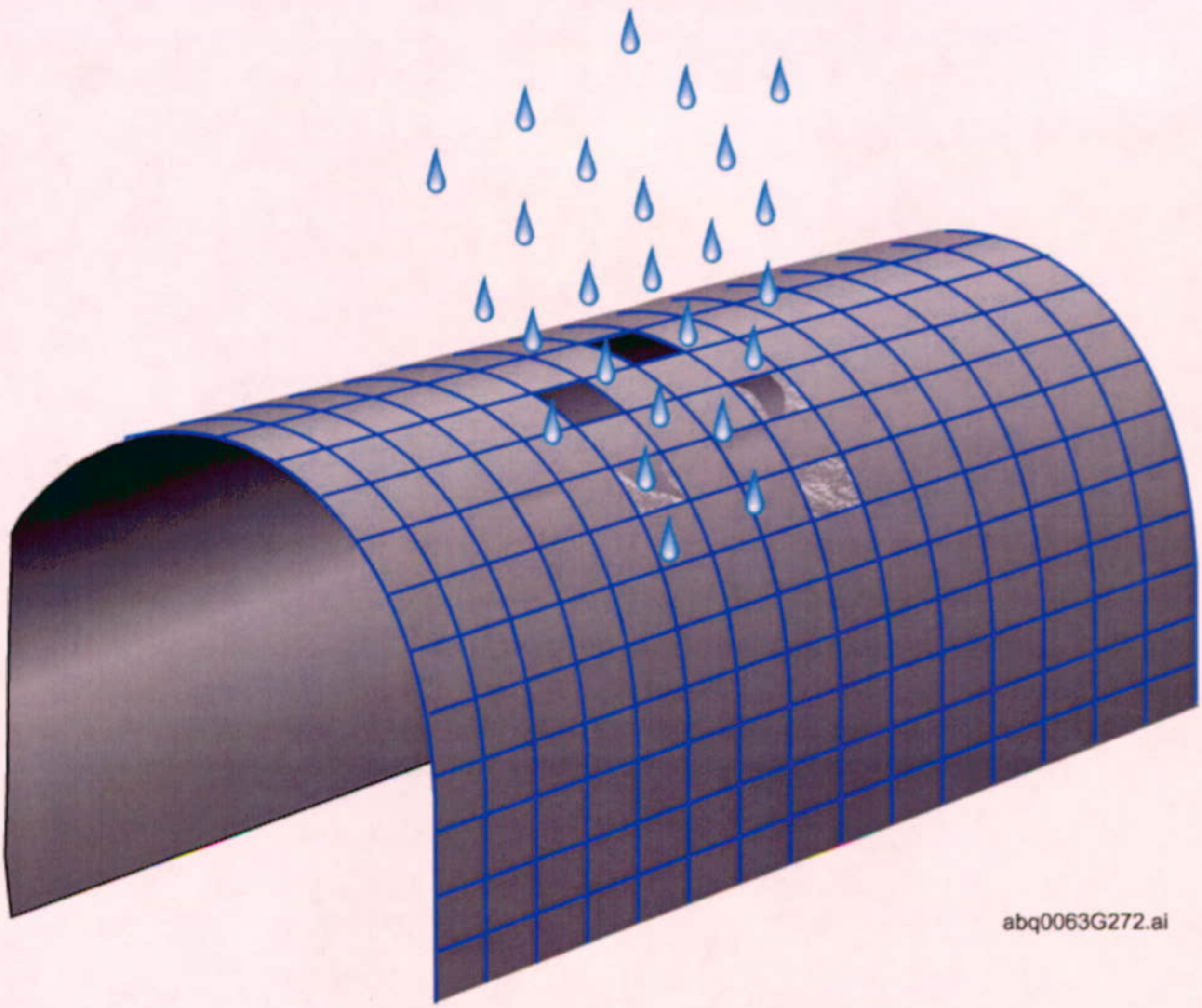
abq0063G640

Figure 3.4-15. Variability CDFs for Alloy-22 with 75, 50, and 25 Percent Variability using 25th Uncertainty Quantile



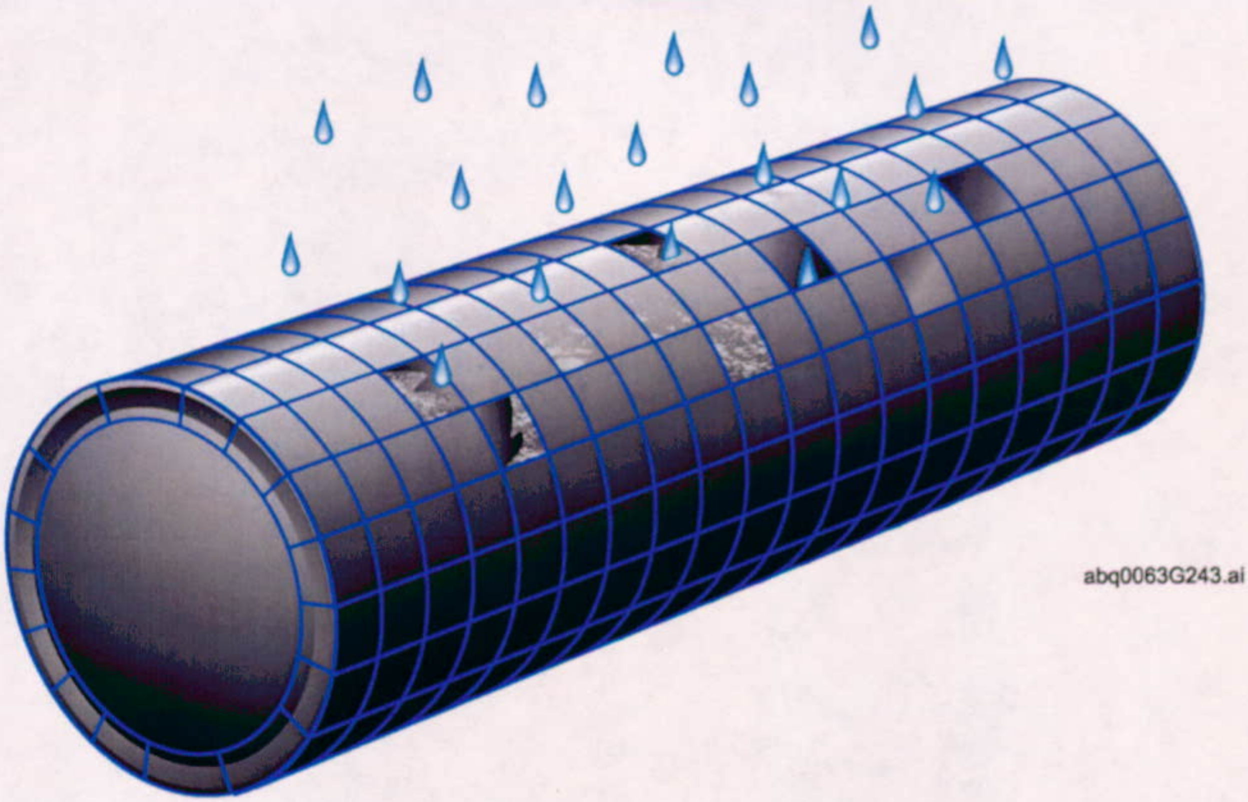
abq0063G641

Figure 3.4-16. Variability CDFs for Alloy-22 with 75, 50, and 25 Percent Variability using 75th Uncertainty Quantile



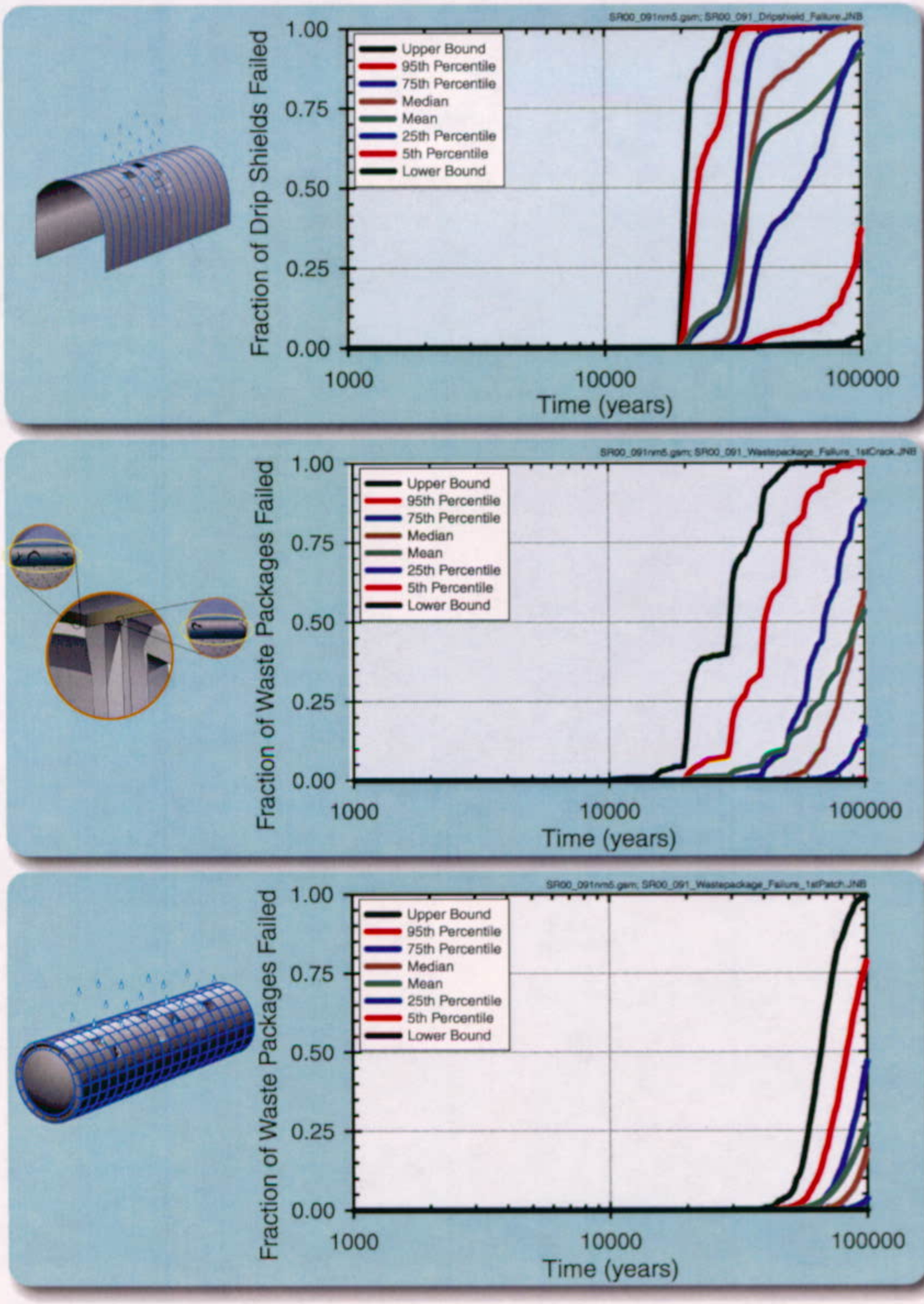
abq0063G272

Figure 3.4-17. Schematic of Drip Shield Implementation in WAPDEG



abq0063G243

Figure 3.4-18. Schematic of Waste Package Implementation in WAPDEG

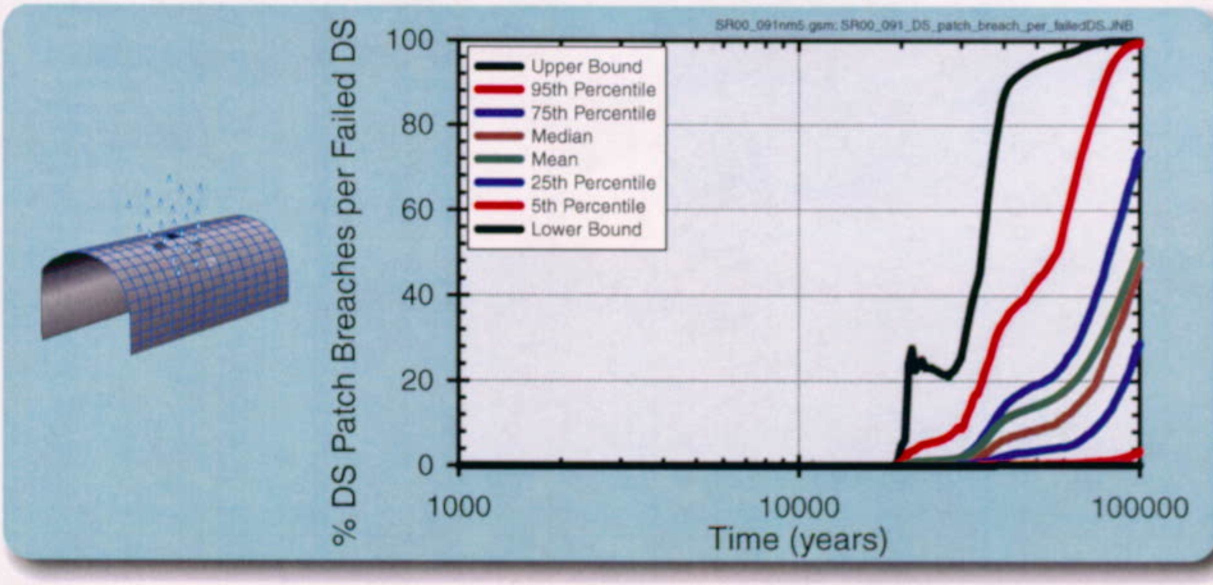


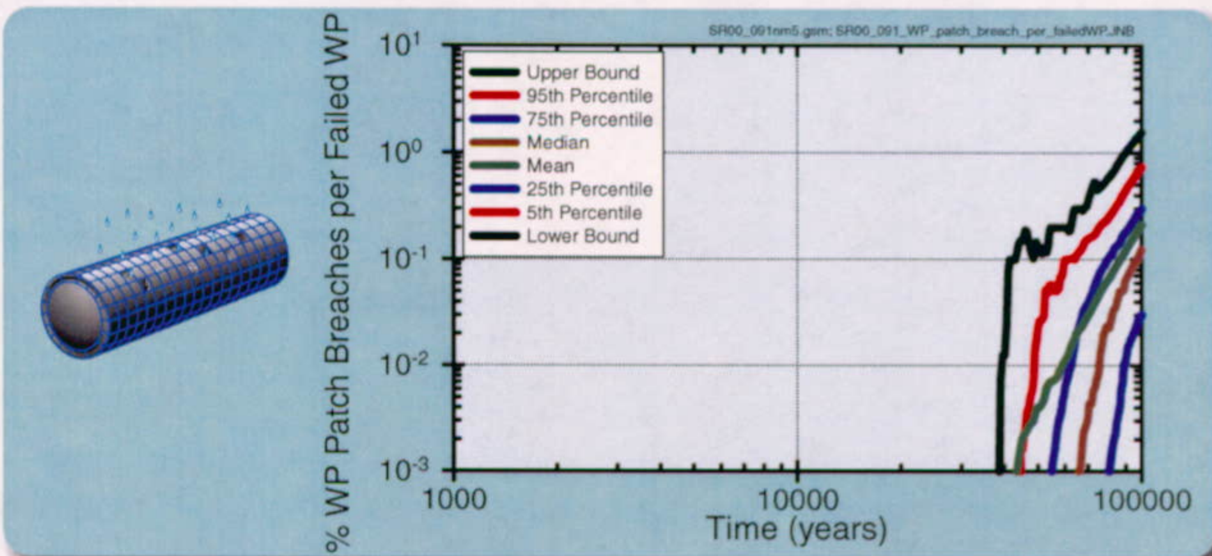
abq0063G273

DTN: MO0010MWDWAP01.009 [153127]

abq0063G273

Figure 3.4-19. Degradation Profiles for Time to First Failure: Drip Shield Patch, Waste Package Crack, Waste Package Patch

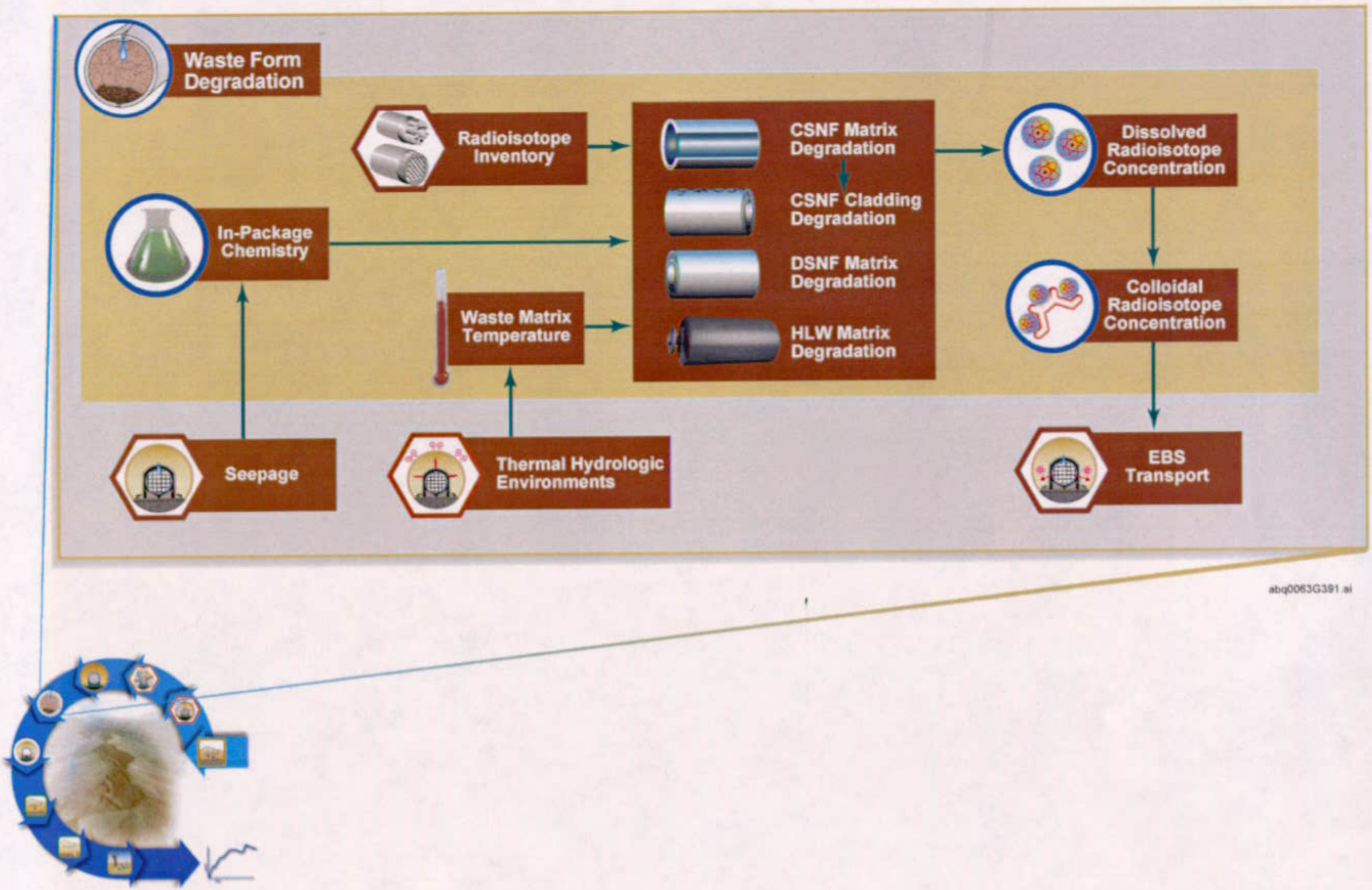




DTN: MO0010MWDWAP01.009 [153127]

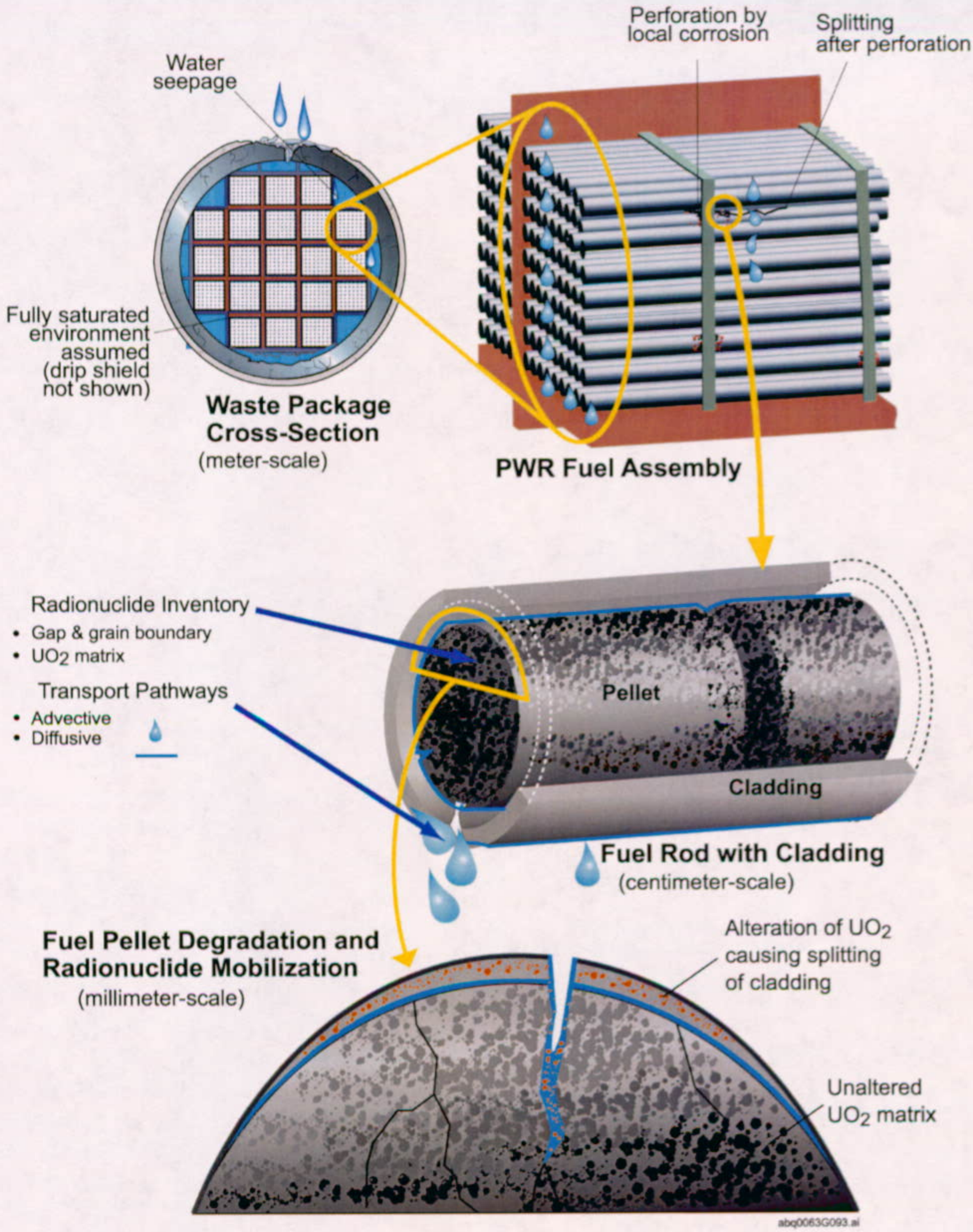
abq0063G415

Figure 3.4-20. Degradation Profiles for Percentage of Patch Failures on Failed Drip Shields and Waste Packages



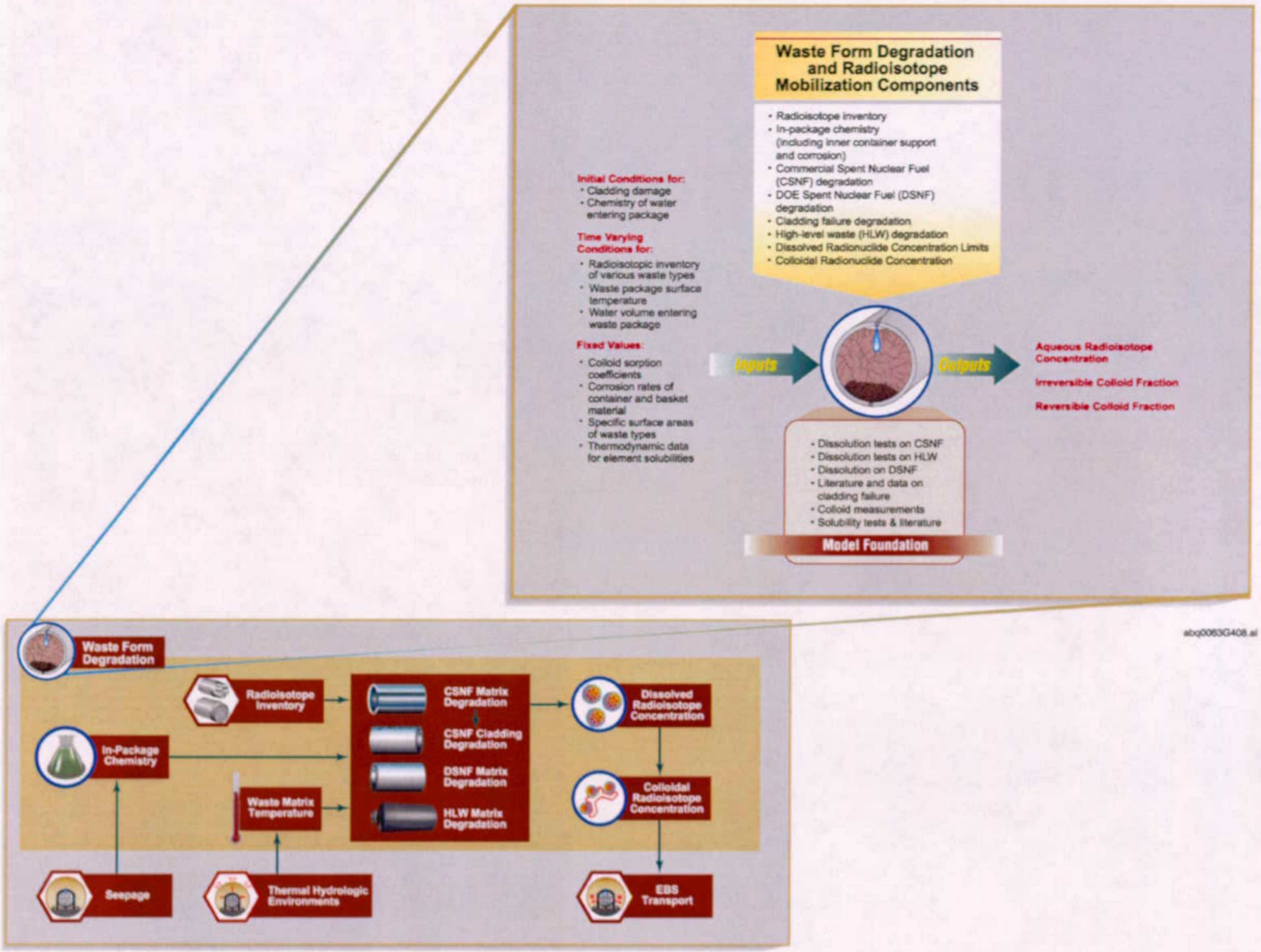
abq0063G391

Figure 3.5-1. Conceptual Model of In-Package Chemistry



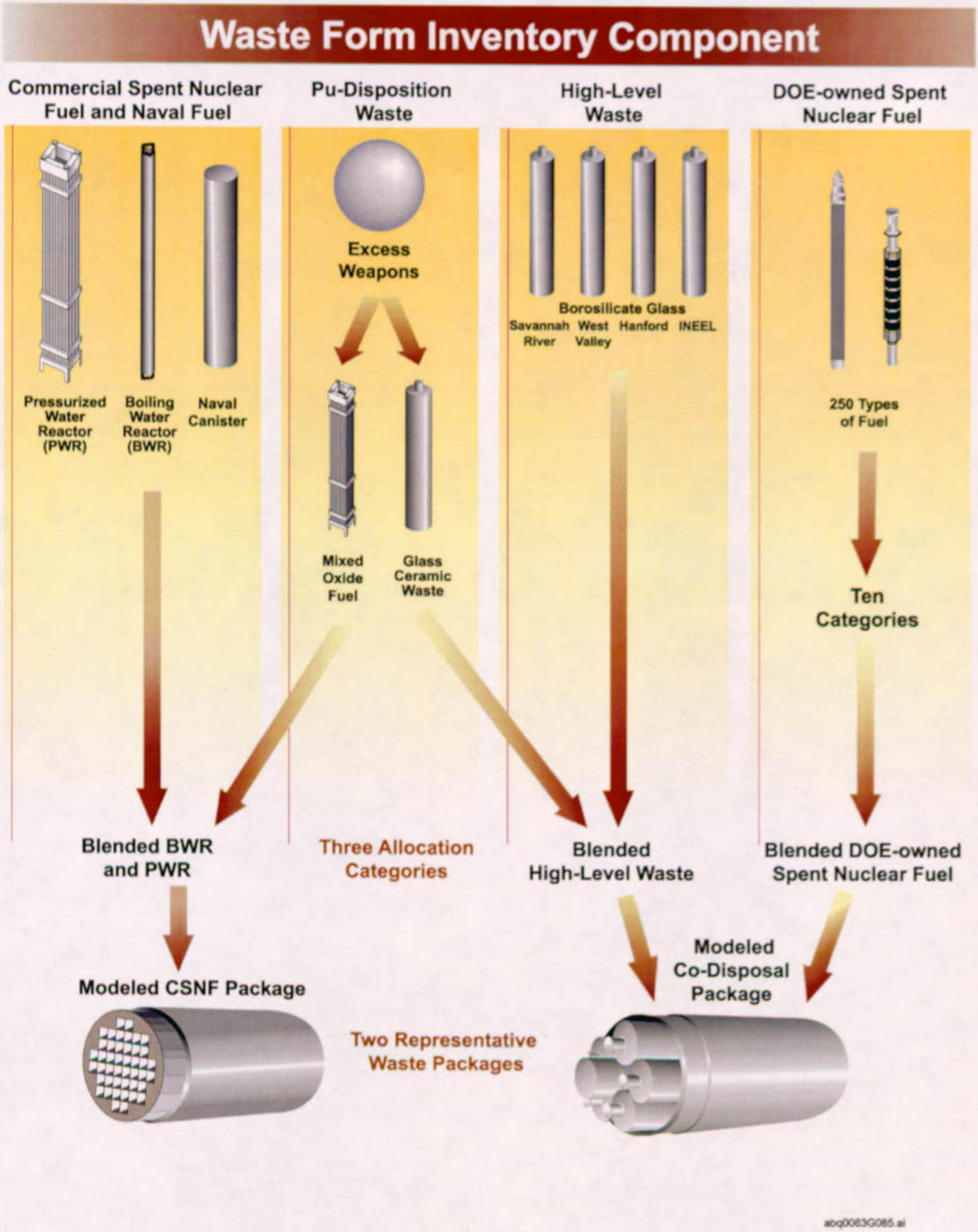
abq0063G093

Figure 3.5-2. Schematic of Waste Form and Waste Package Degradation Mechanisms



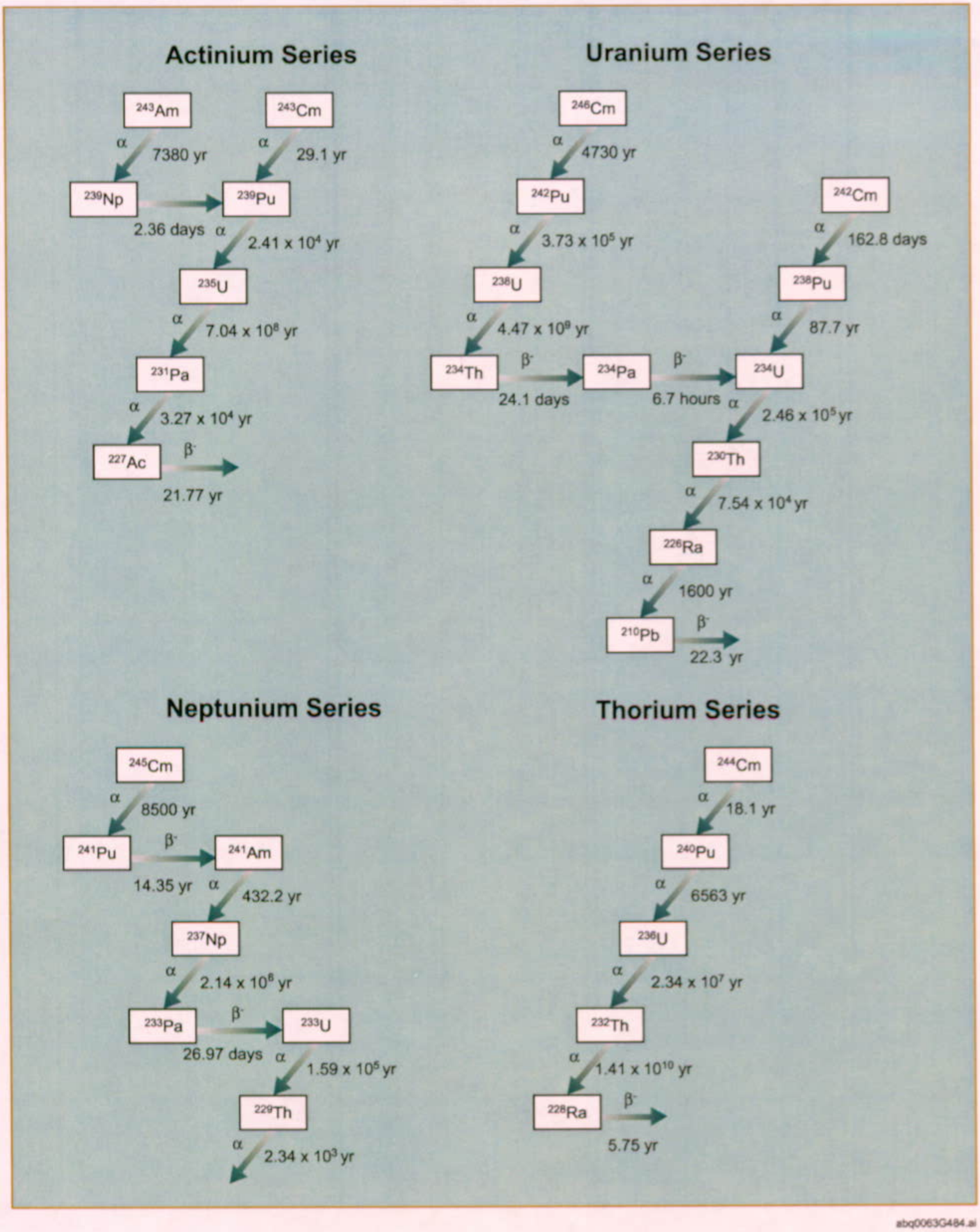
abq0063G408

Figure 3.5-3. Summary of Inputs, Outputs, Components, and Assumptions of Waste Form Degradation Model



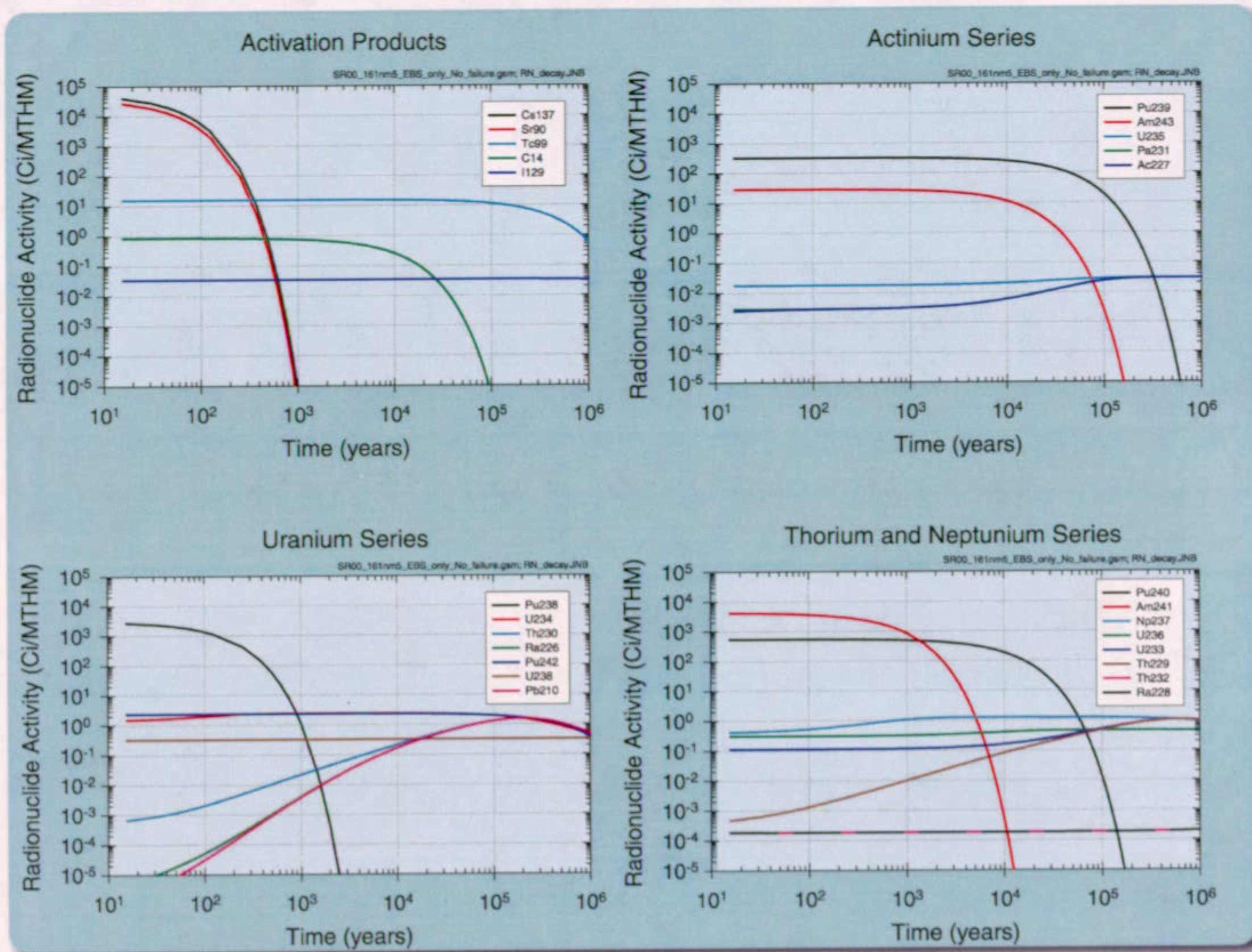
abq0063G085

Figure 3.5-4. Waste Types Grouped into Three Waste Allocation Categories and Two Representative Waste Packages for Modeling in Total System Performance Assessment-Site Recommendation Analysis



abq0063G484

Figure 3.5-5. Decay Chains of the Actinide Elements



abq0063G486.ai

abq0063G486

Figure 3.5-6. Decay History for the Products and Actinide Elements for 1,000,000-year Time Period Activation (a) Activation Products, (b) Actinium Series, (c) Uranium Series, and (d) Thorium and Uranium Series

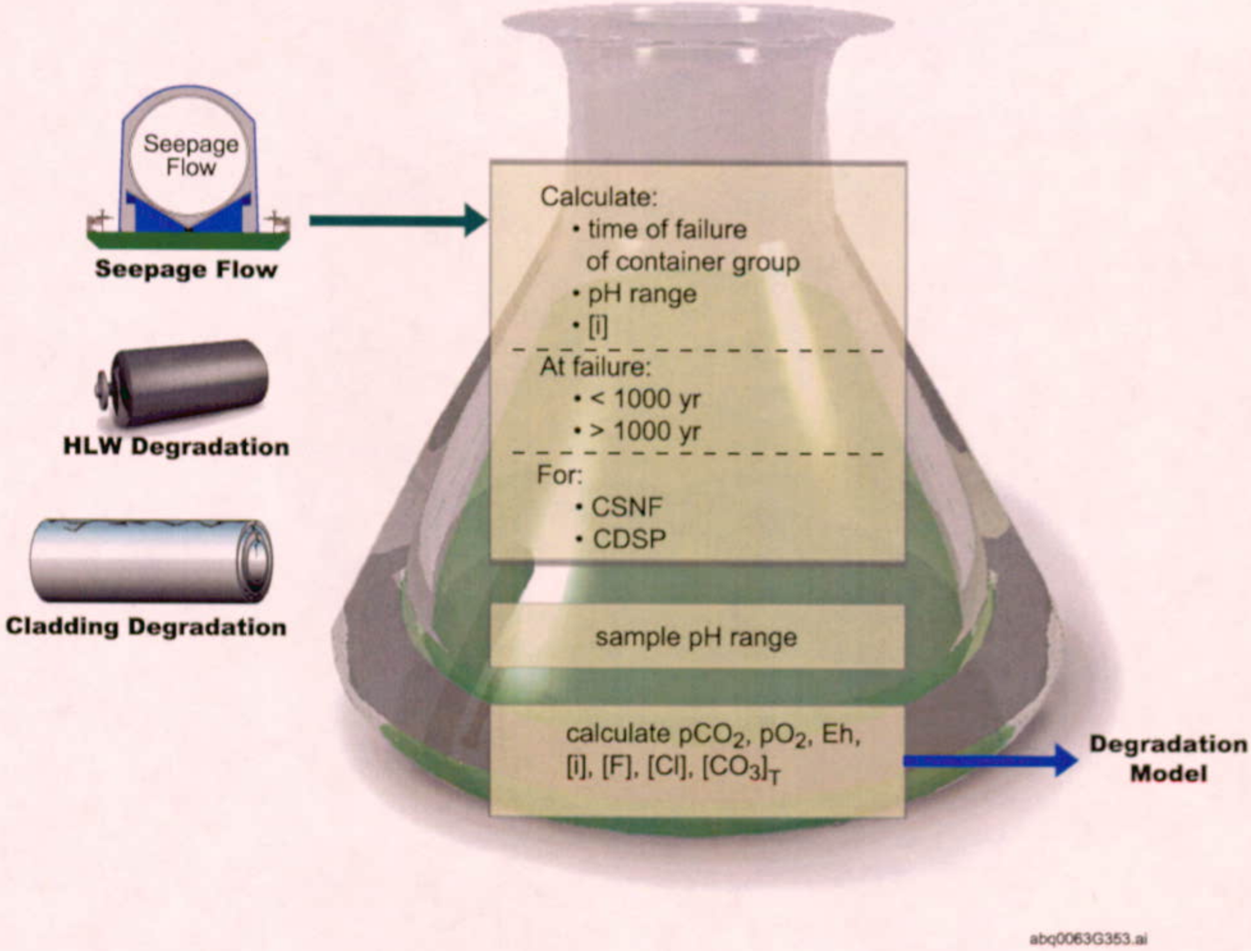


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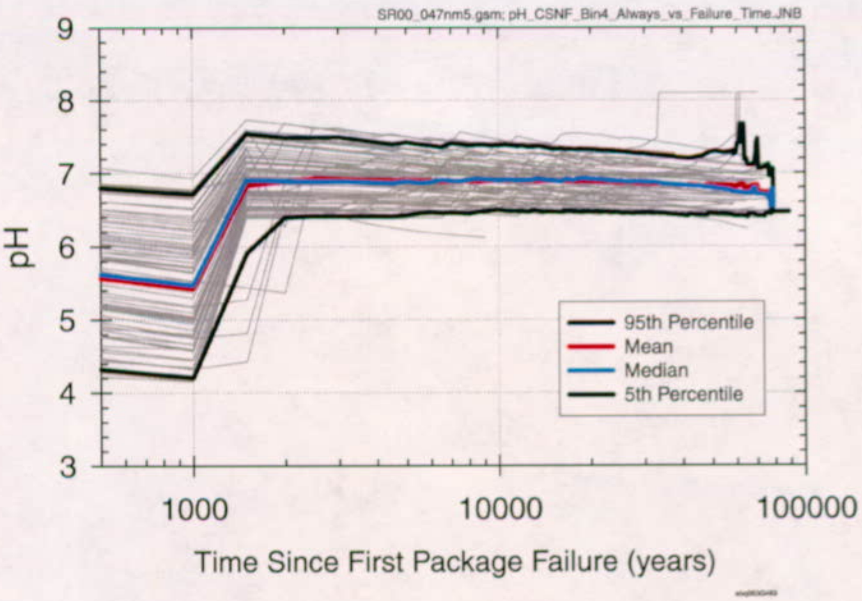
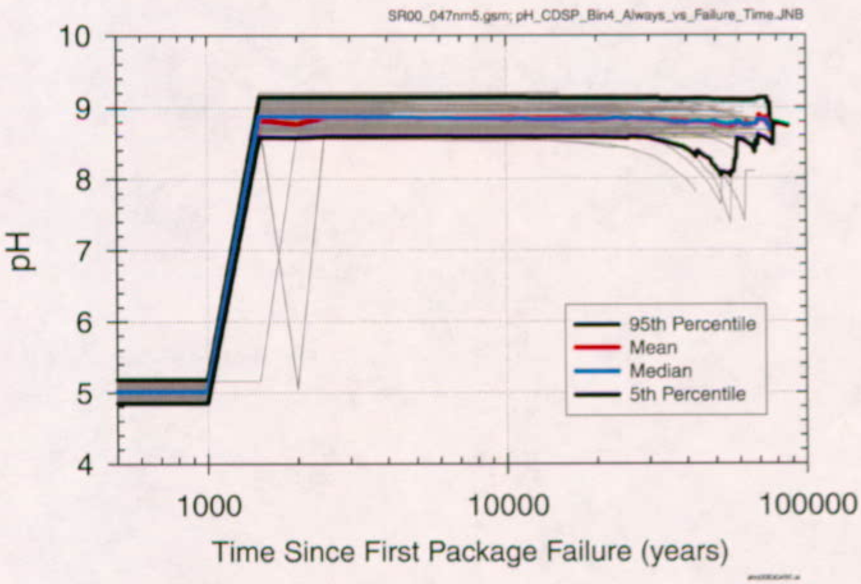
Figure 3.5-7. Implementation of the In-Package Chemistry Component



(a) Commercial Spent Nuclear Fuel Packages

abq0063G489



(b) Co-Disposal Packages

abq0063G490

Figure 3.5-8. pH of Packages in 20 to 60 mm/yr Infiltration Bin versus Time since Failure of Waste Package (a) Commercial Spent Nuclear Fuel Packages, (b) Co-Disposal Packages

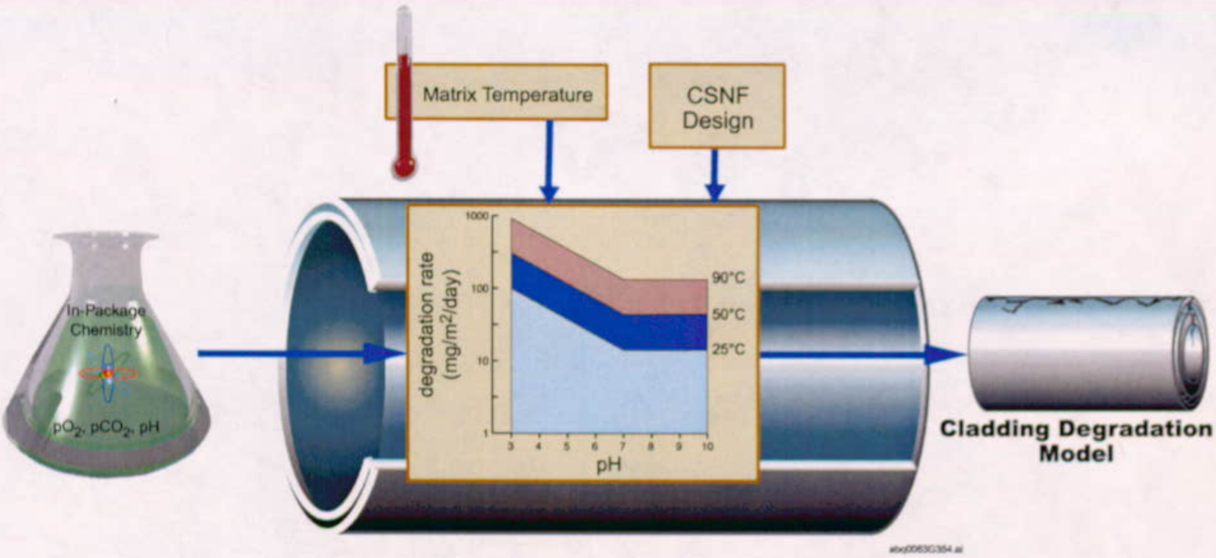


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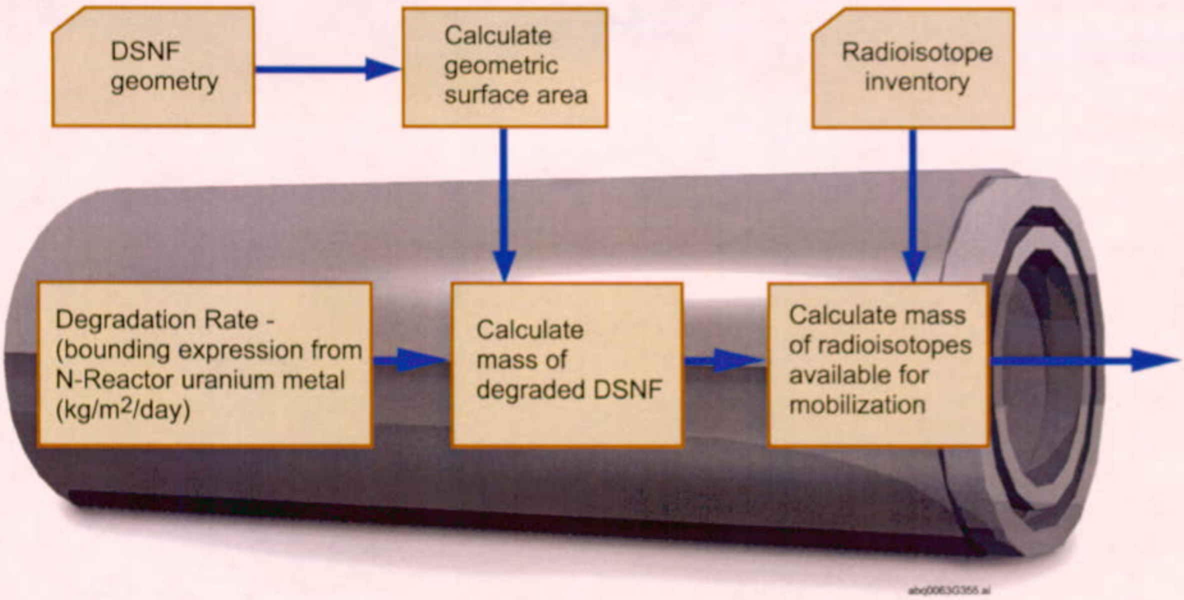
Figure 3.5-9. Commercial Spent Nuclear Fuel Matrix Degradation Model



abq0063G355

Figure 3.5-10. Implementation of DOE-Owned Spent Nuclear Fuel Degradation Component in Waste Form Degradation Model

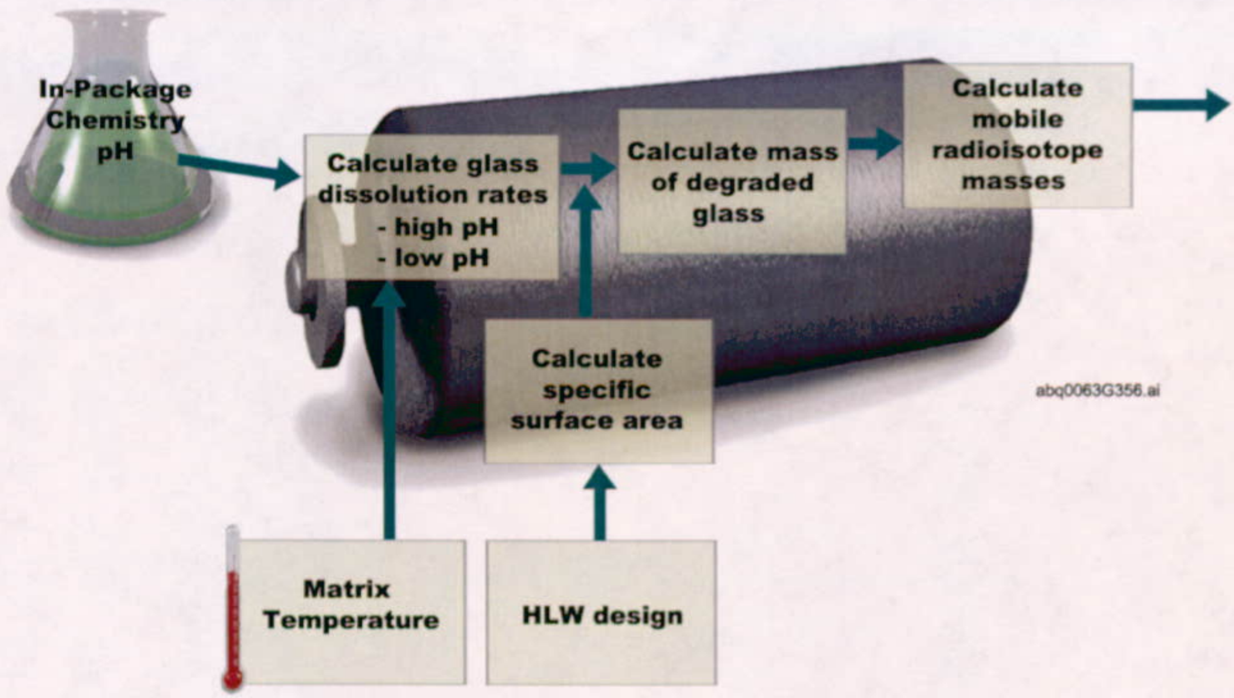


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Figure 3.5-11. High-Level Waste Degradation Component

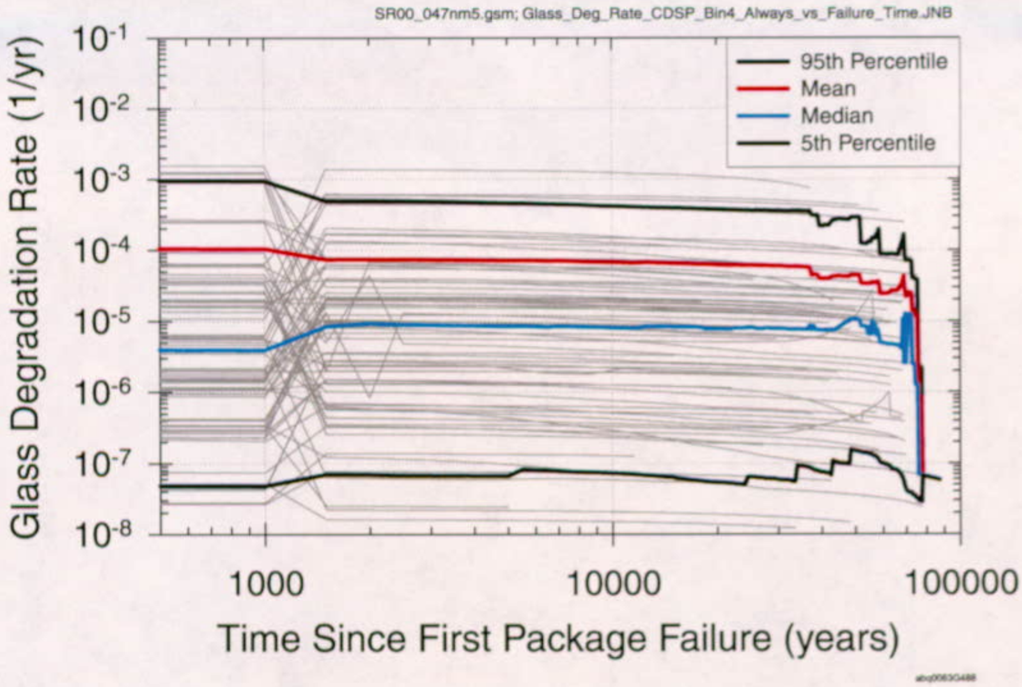

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Figure 3.5-12. Range of Glass Degradation Rates Calculated for 20 to 60 mm/yr Infiltration Bin for always Drip Condition versus Time since Waste Package First Perforated

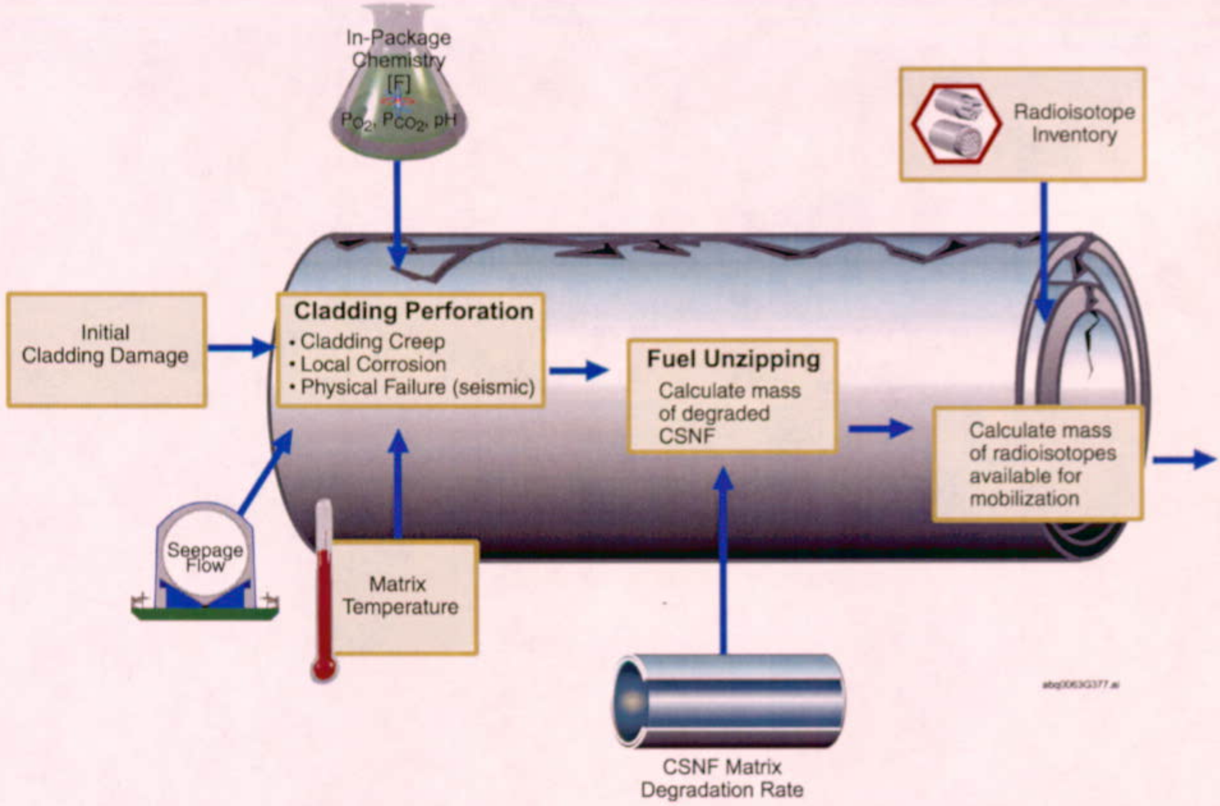

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Figure 3.5-13. Implementation of Commercial Spent Nuclear Fuel Cladding Degradation Component in Waste Form Degradation Model

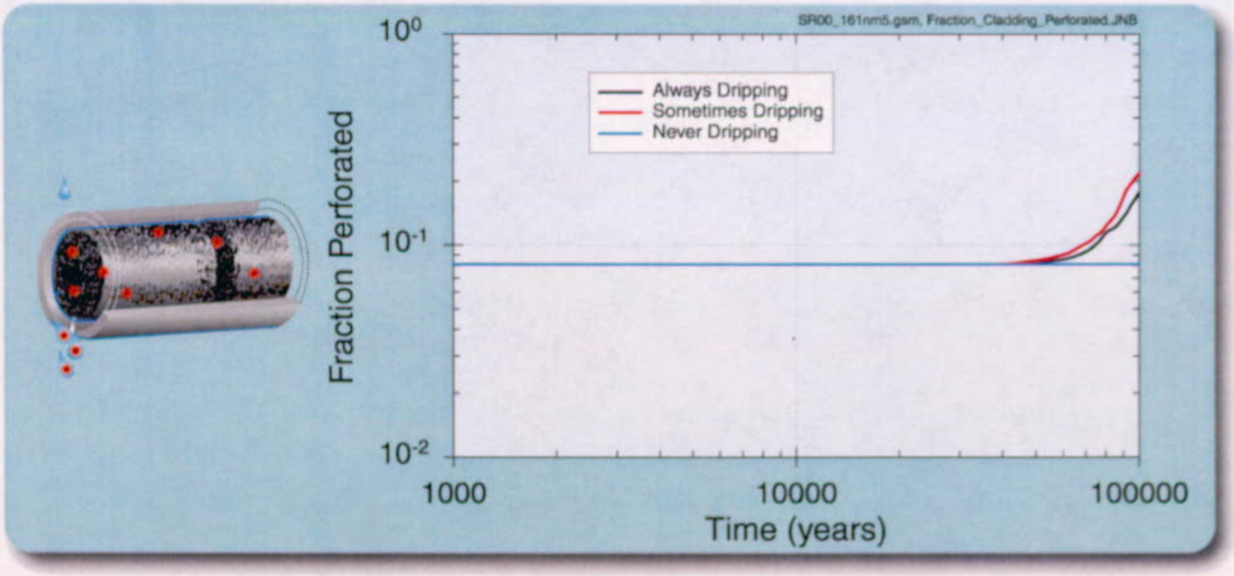

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Figure 3.5-14. Mean Fraction of Cladding Perforated for 20 to 60 mm/yr Infiltration Bin for all Three Drip Conditions

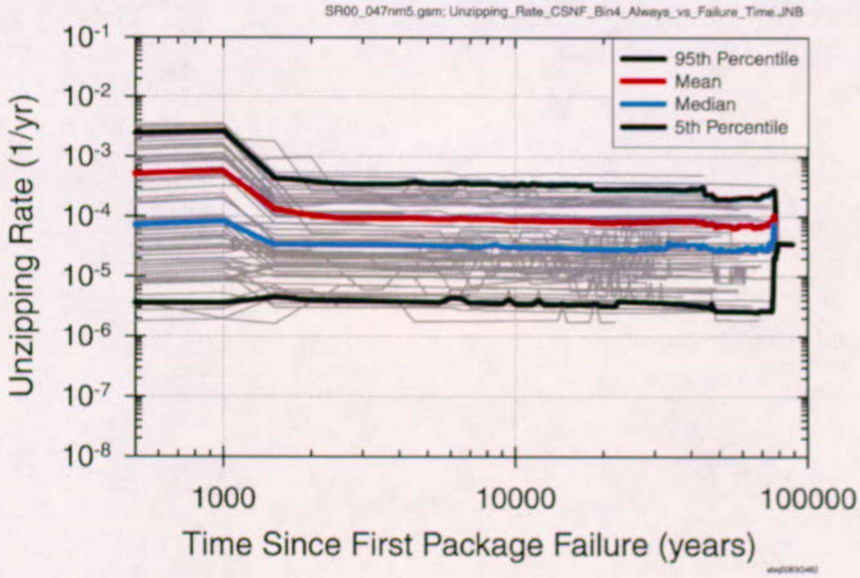

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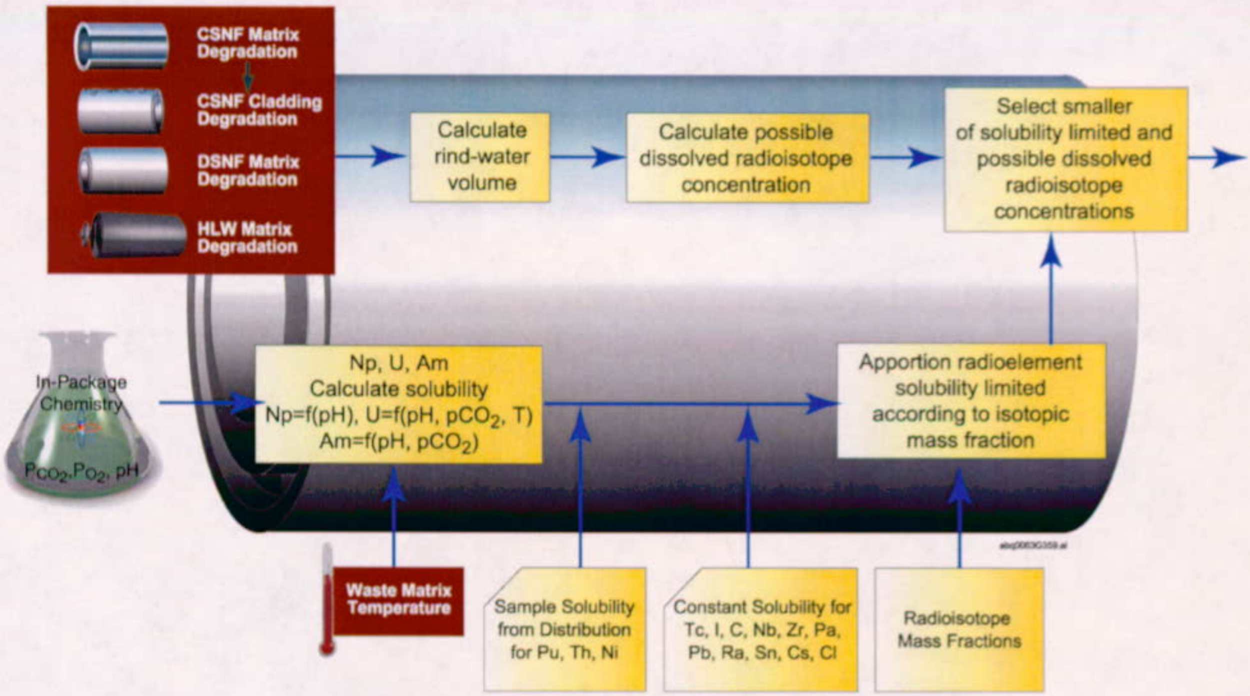
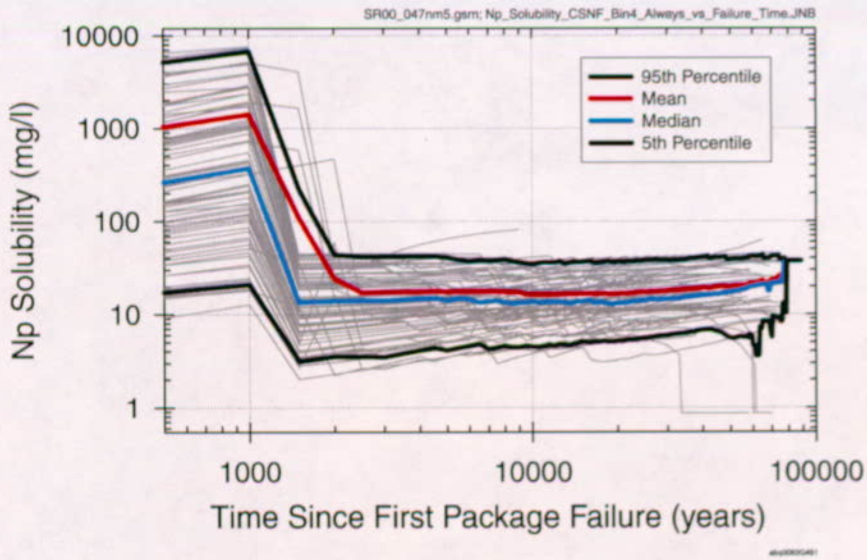
Figure 3.5-15. Mean Unzipping Rate for Commercial Spent Nuclear Fuel Cladding Calculated for 20 to 60 mm/yr Infiltration Bin for always Drip Condition versus Time since Waste Package First Perforated



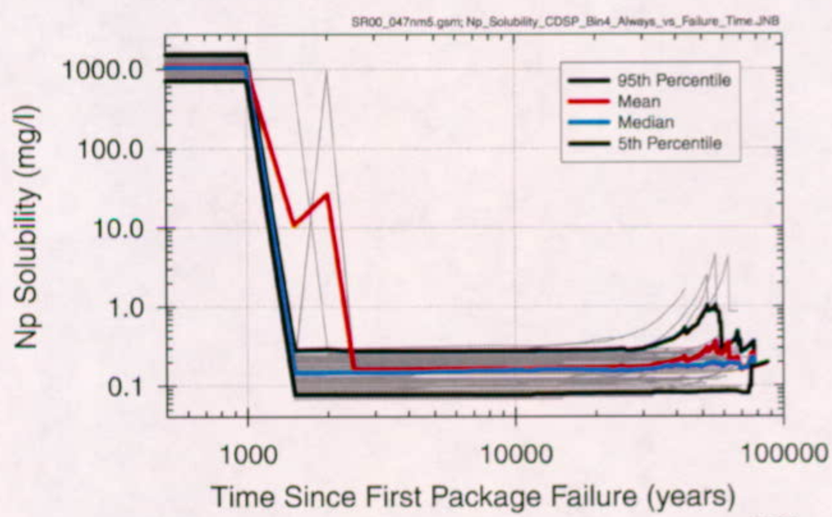
abq0063G359

Figure 3.5-16. Implementation of Solubility Component in Waste Form Degradation Model



abq0063G491

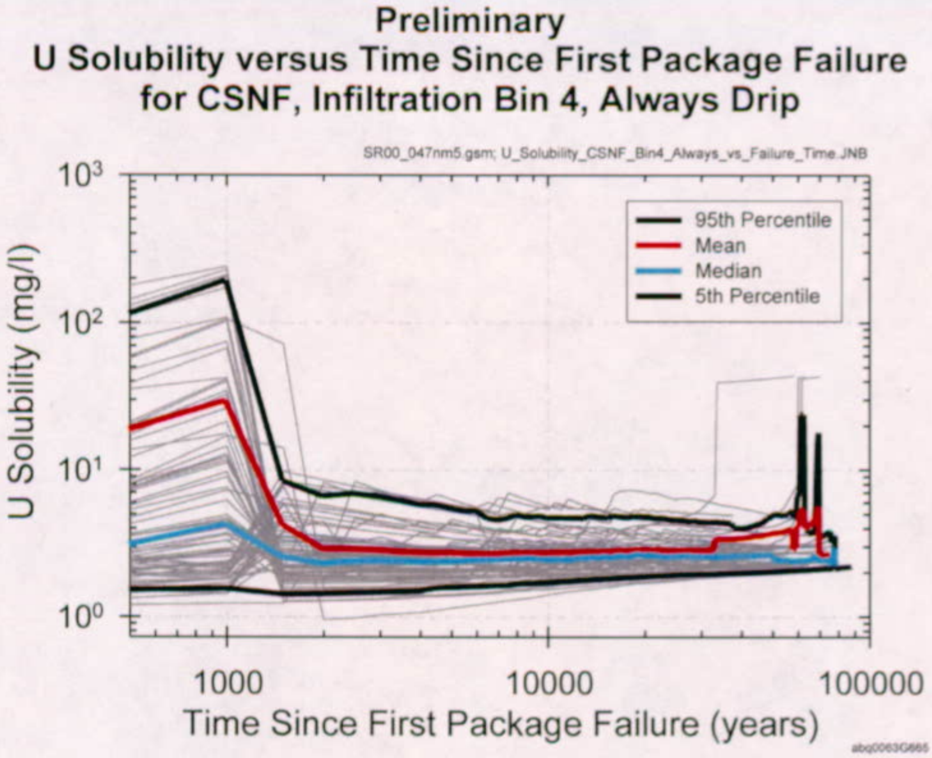
a) CSNF Packages



abq0063G487

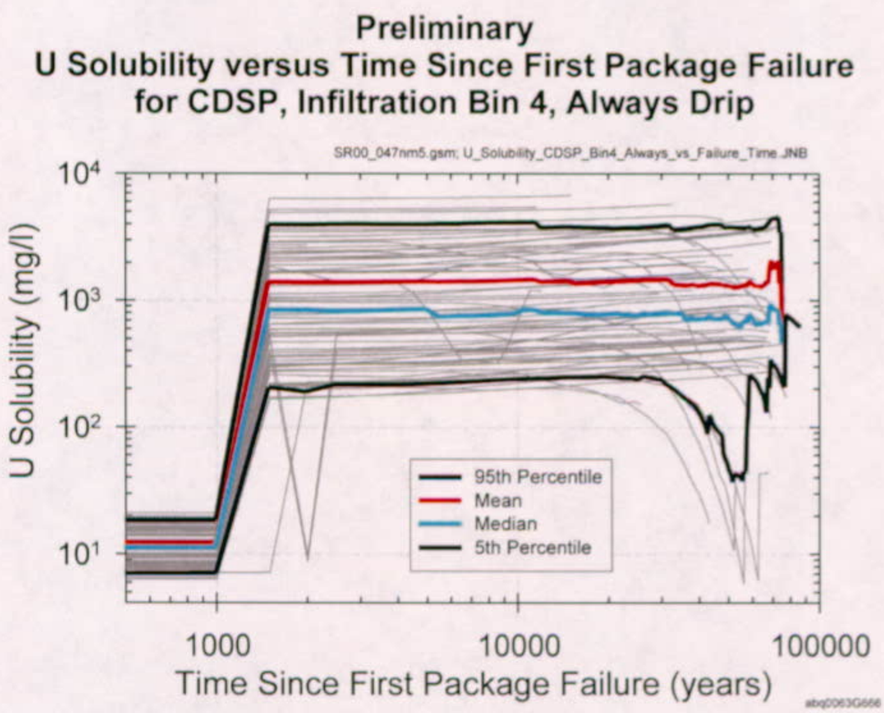
b) Co-disposal Packages

Figure 3.5-17a. Solubility of Np in 20 to 60 mm/yr Infiltration Bin for always Drip Condition versus Time since First Perforation of the Waste Packages (a) CSNF Packages, (b) Co-disposal Packages



c) CSNF Packages

abq0063G665



d) Co-disposal Packages

abq0063G666

Figure 3.5-17b. Solubility of U in 20 to 60 mm/yr Infiltration Bin for always Drip Condition versus Time since First Perforation of the Waste Packages (c) CSNF Packages, (d) Co-disposal Packages

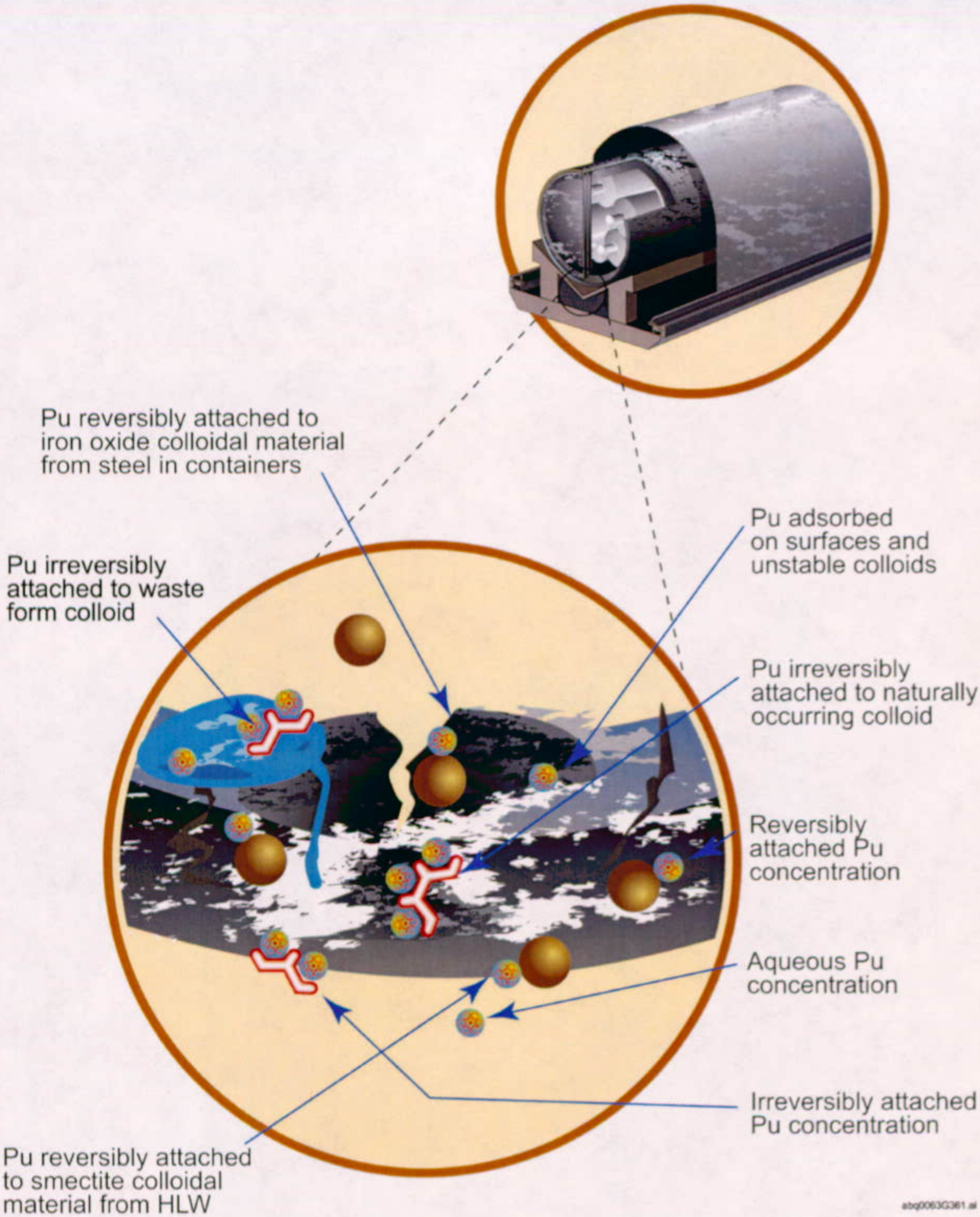


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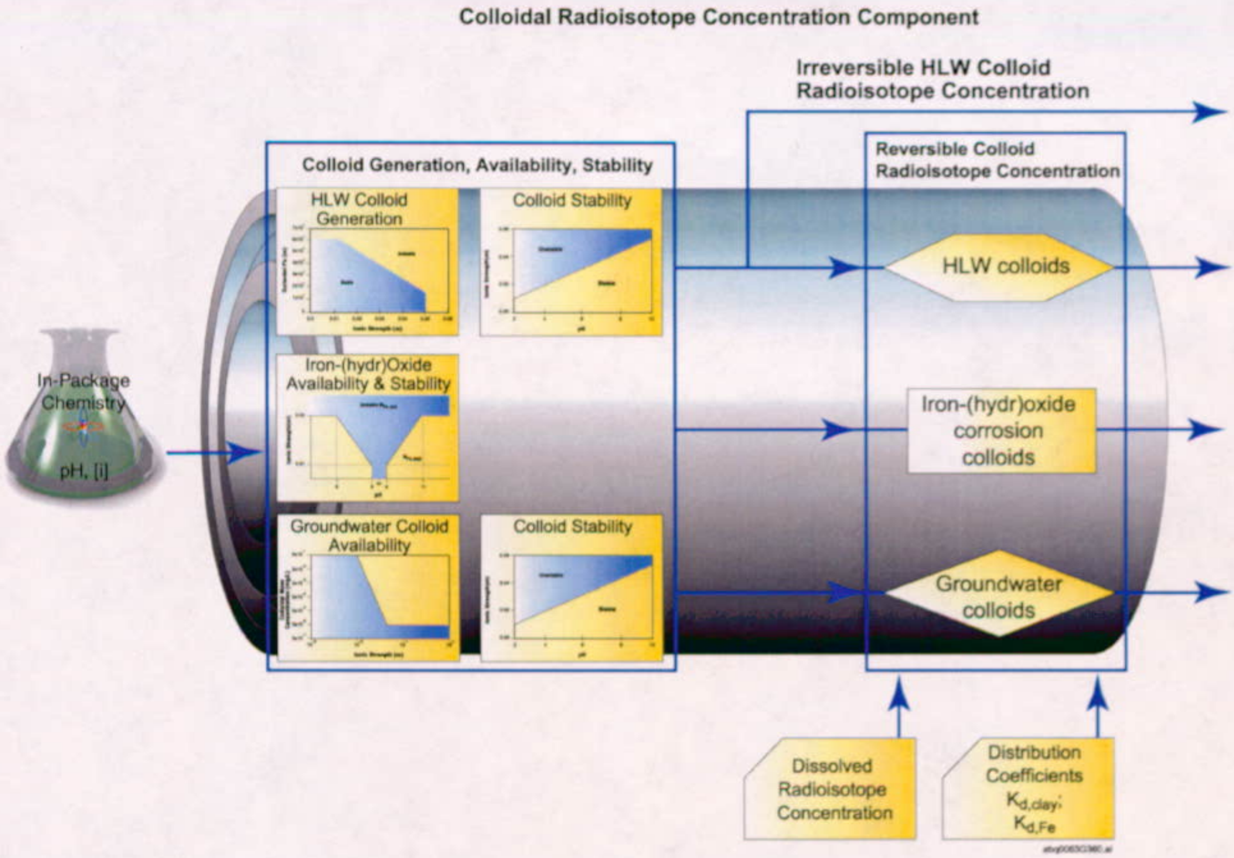
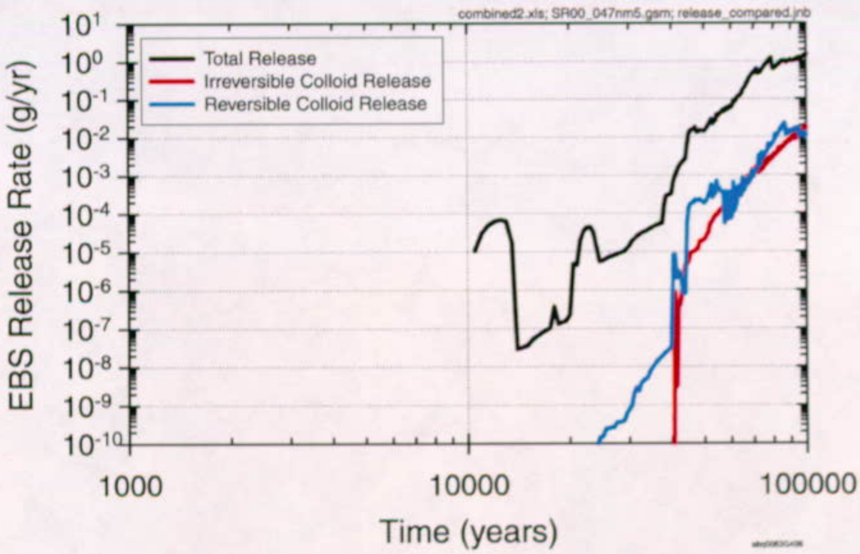
Figure 3.5-18. Conceptual Model of Formation of Reversibly and Irreversibly Attached Radioisotopes on Colloids



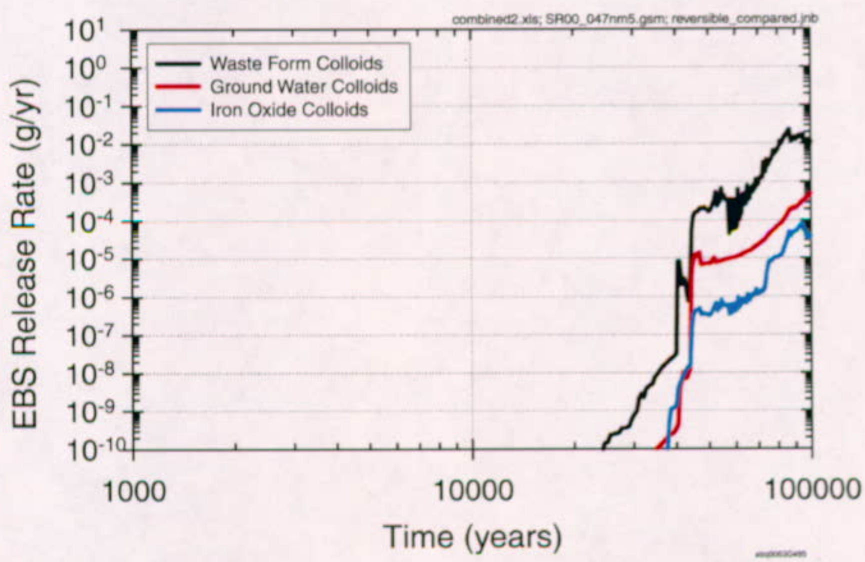
abq0063G360

Figure 3.5-19. Implementation of Colloidal Radioisotope Component in Waste Form Degradation Model



a) Total Release

abq0063G496



b) Source of Reversible Colloids

abq0063G495

Figure 3.5-20. Contribution of Colloids to Release of ^{239}Pu (a) Total Release, (b) Source of Reversible Colloids

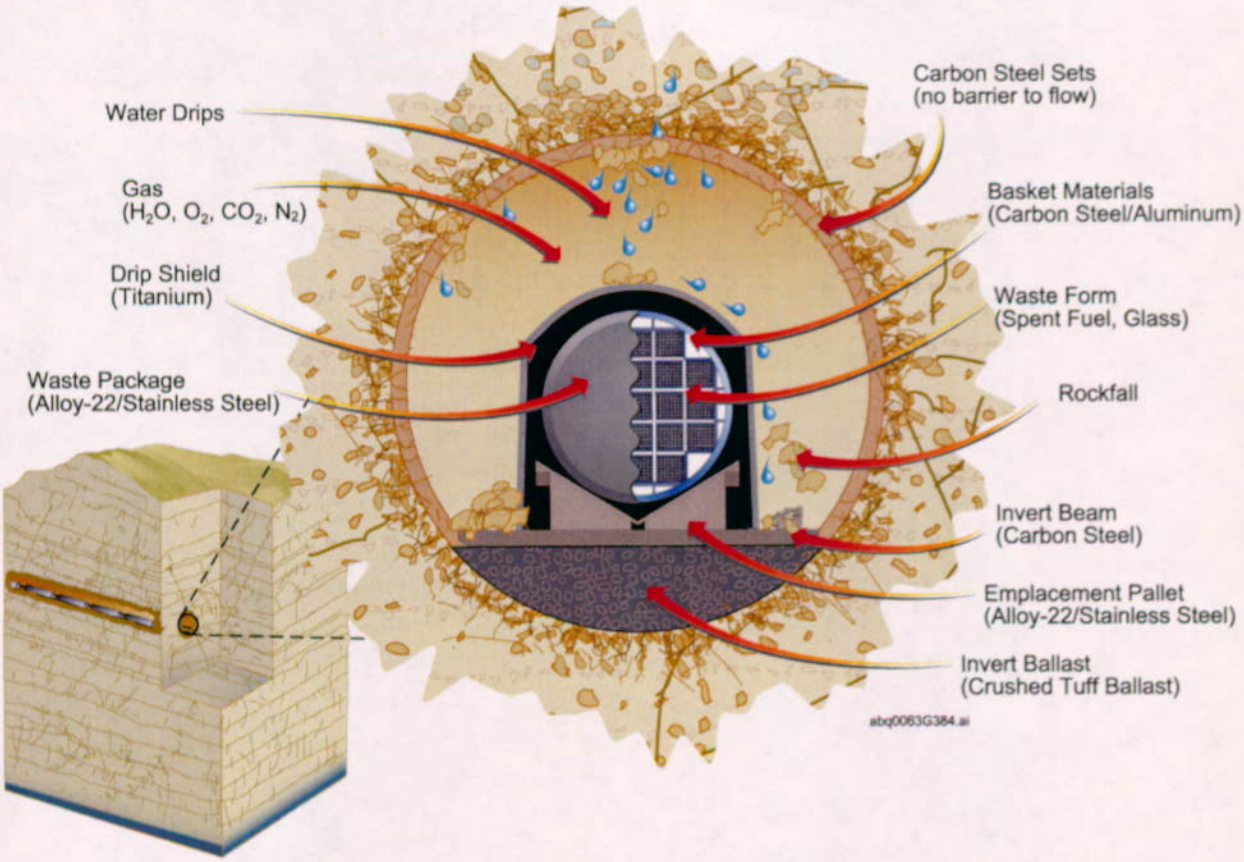


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Figure 3.6-1. Cross-Section of a Typical Emplacement Drift Showing the Major Components of the Engineered Barrier System

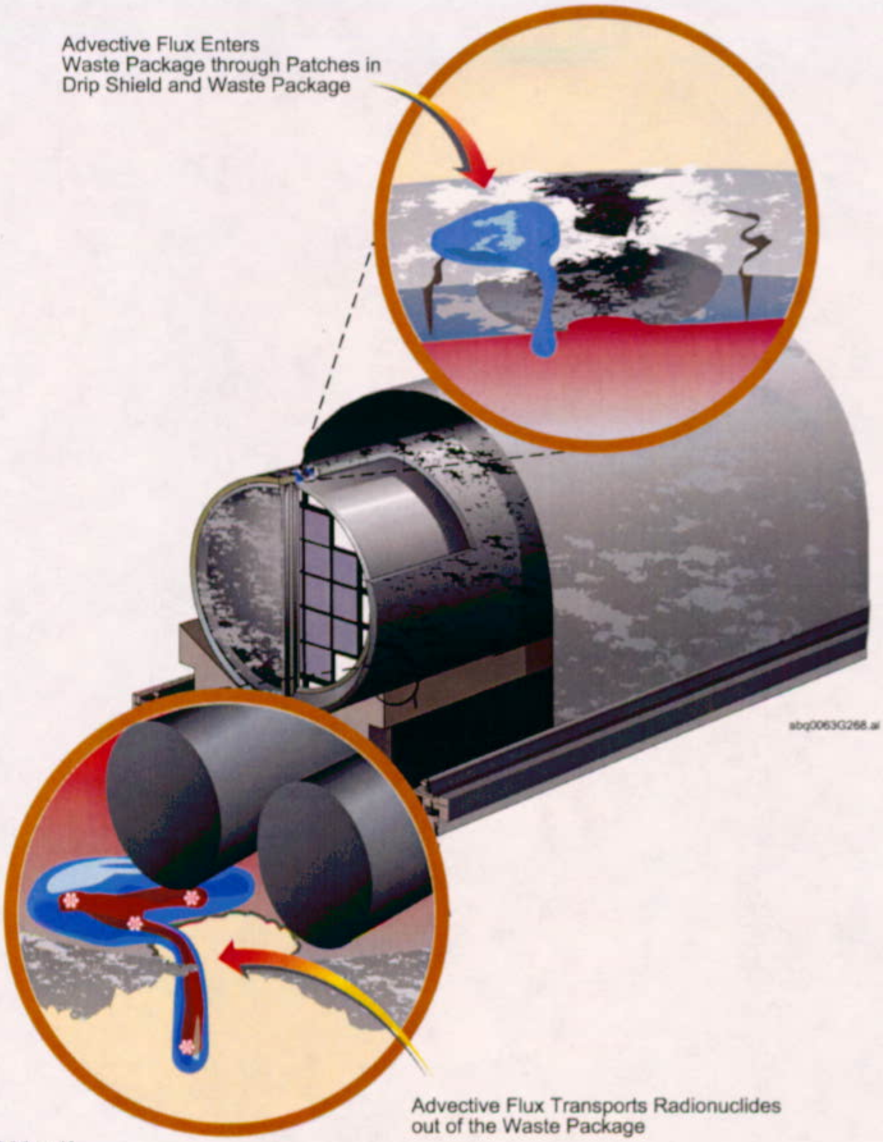

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Figure 3.6-2. Advective Flux through Patches can Transport Radionuclides out of a Breached Drip Shield and Waste Package

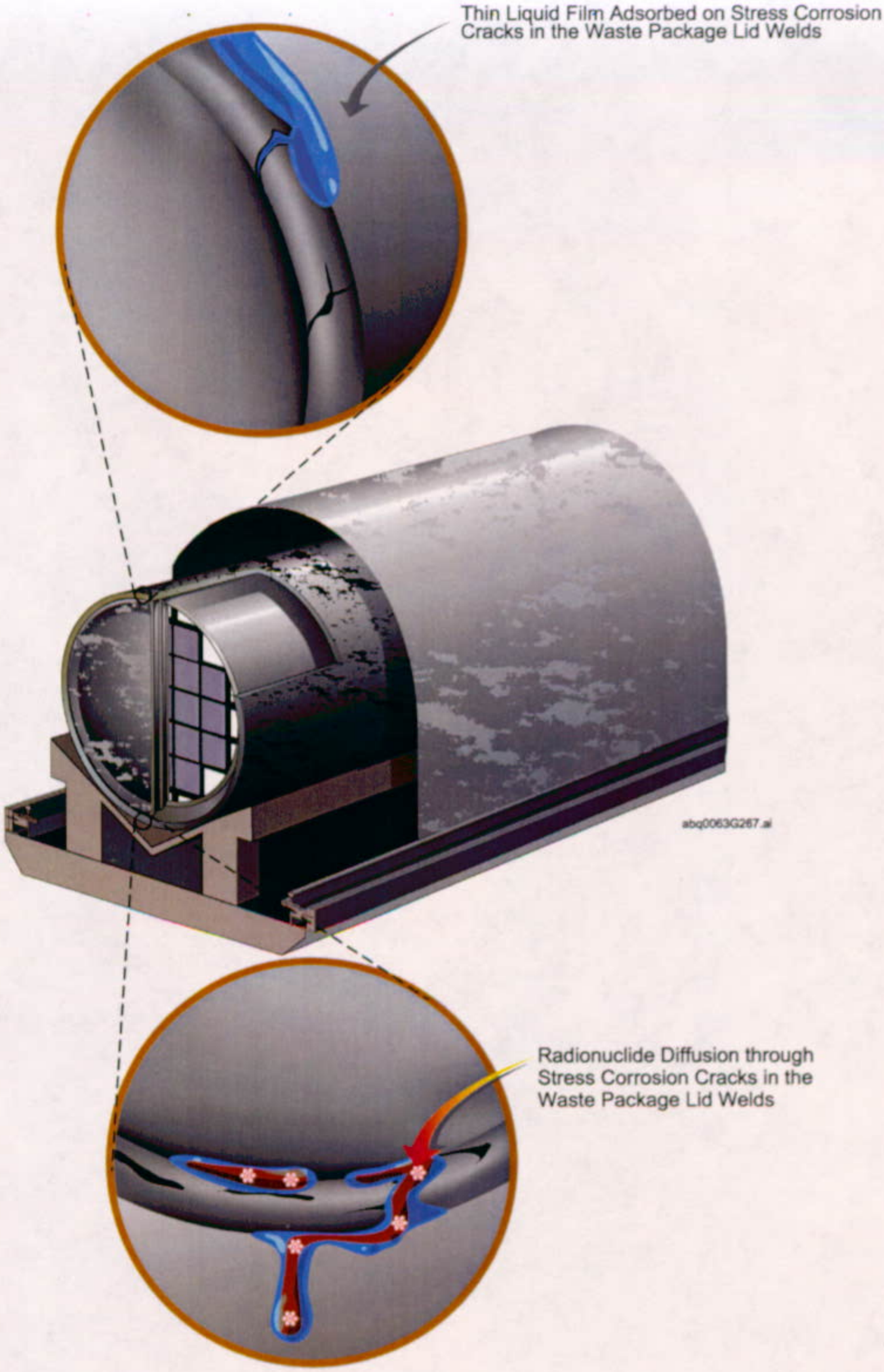

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Figure 3.6-3. Diffusion of Radionuclides through Stress Corrosion Cracks can Transport Radionuclides out of the Waste Package

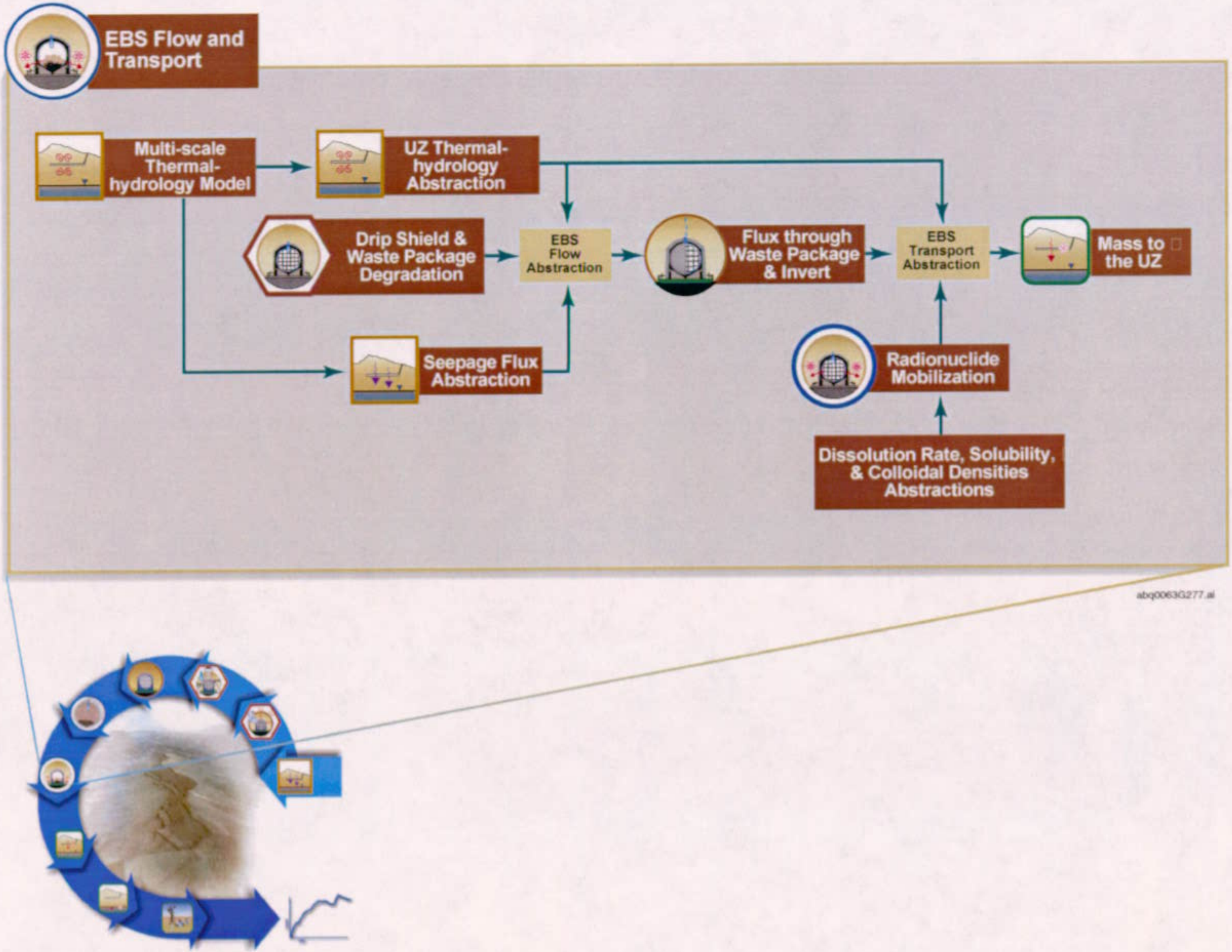

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Figure 3.6-4. The Abstractions for Engineered Barrier System Flow and Engineered Barrier System Transport Require Inputs from Many Elements of the Total System Performance Assessment

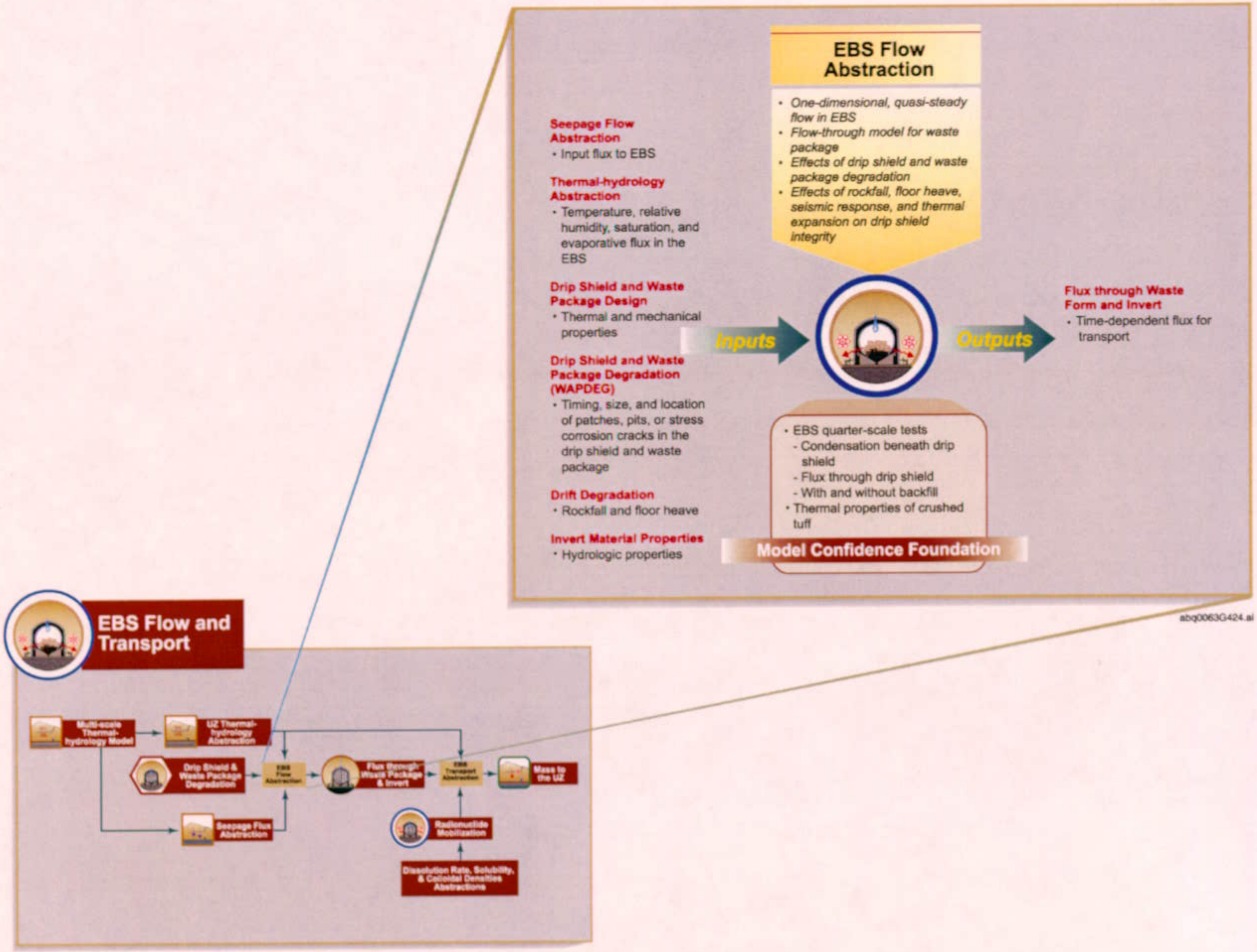

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Figure 3.6-5. Summary of Conceptual Model for Engineered Barrier System Flow Abstraction

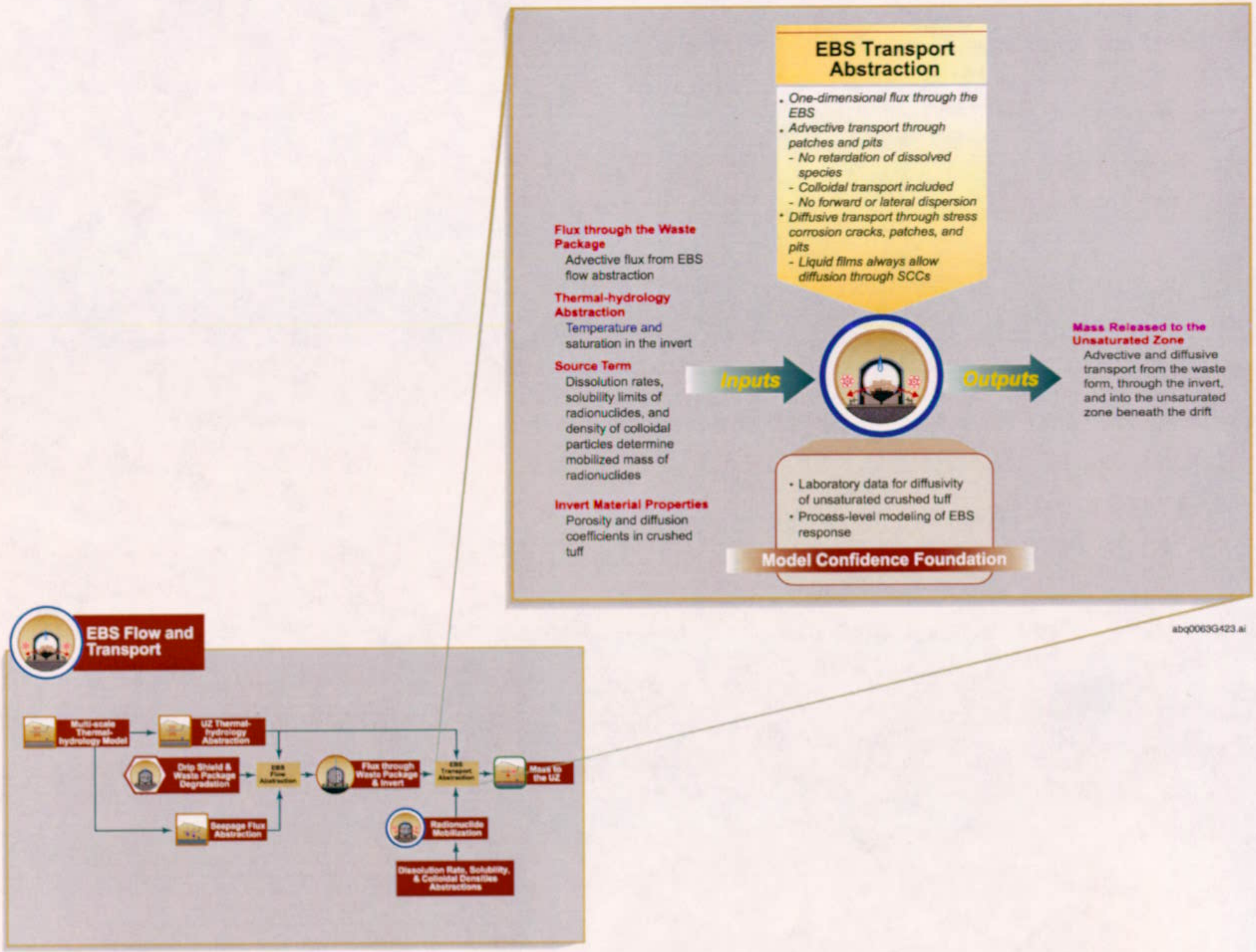

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Figure 3.6-6. Summary of Conceptual Model for Engineered Barrier System Transport Abstraction

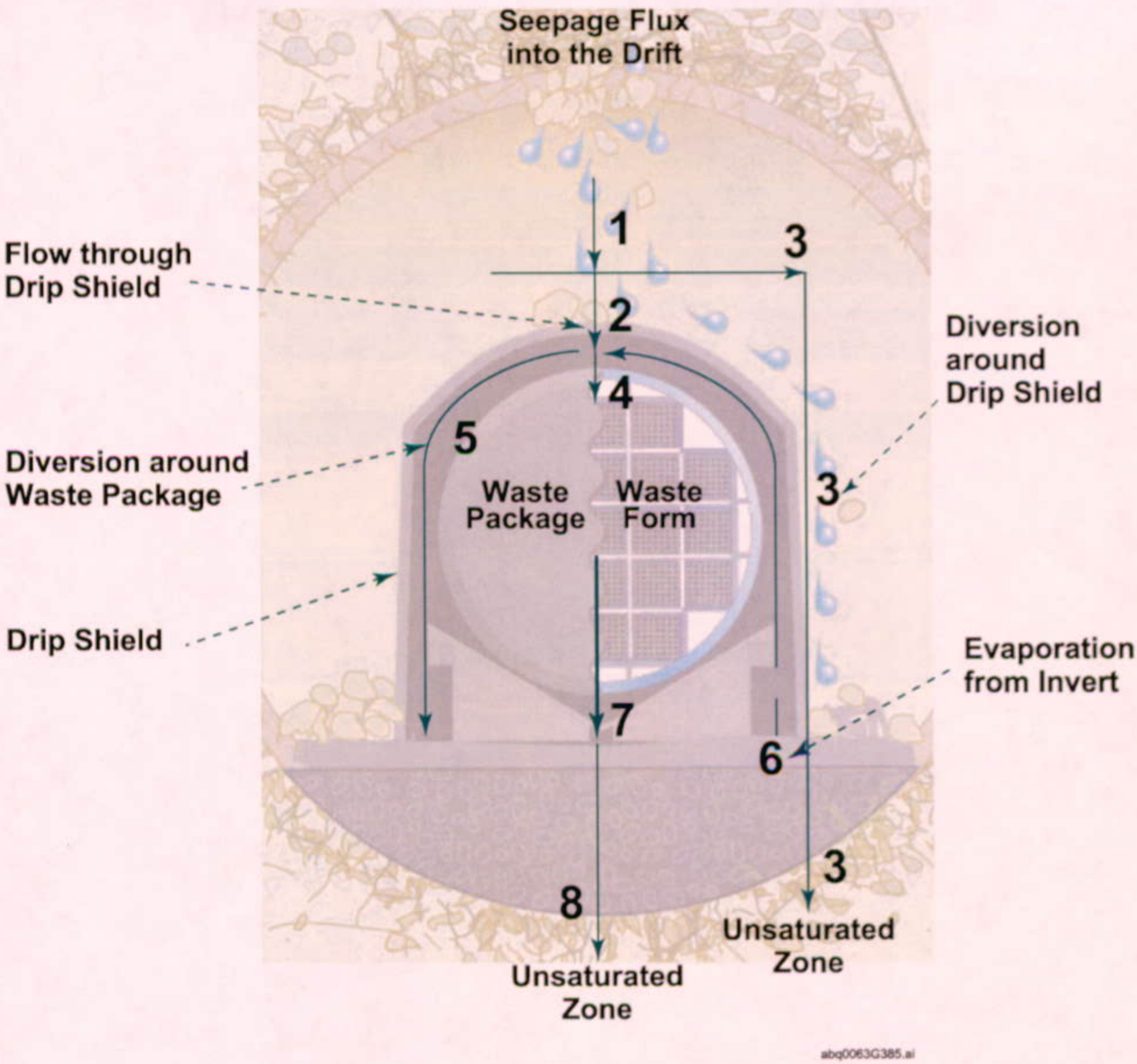

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Figure 3.6-7. Schematic Diagram of the Flow Pathways in the Engineered Barrier System Flow Abstraction

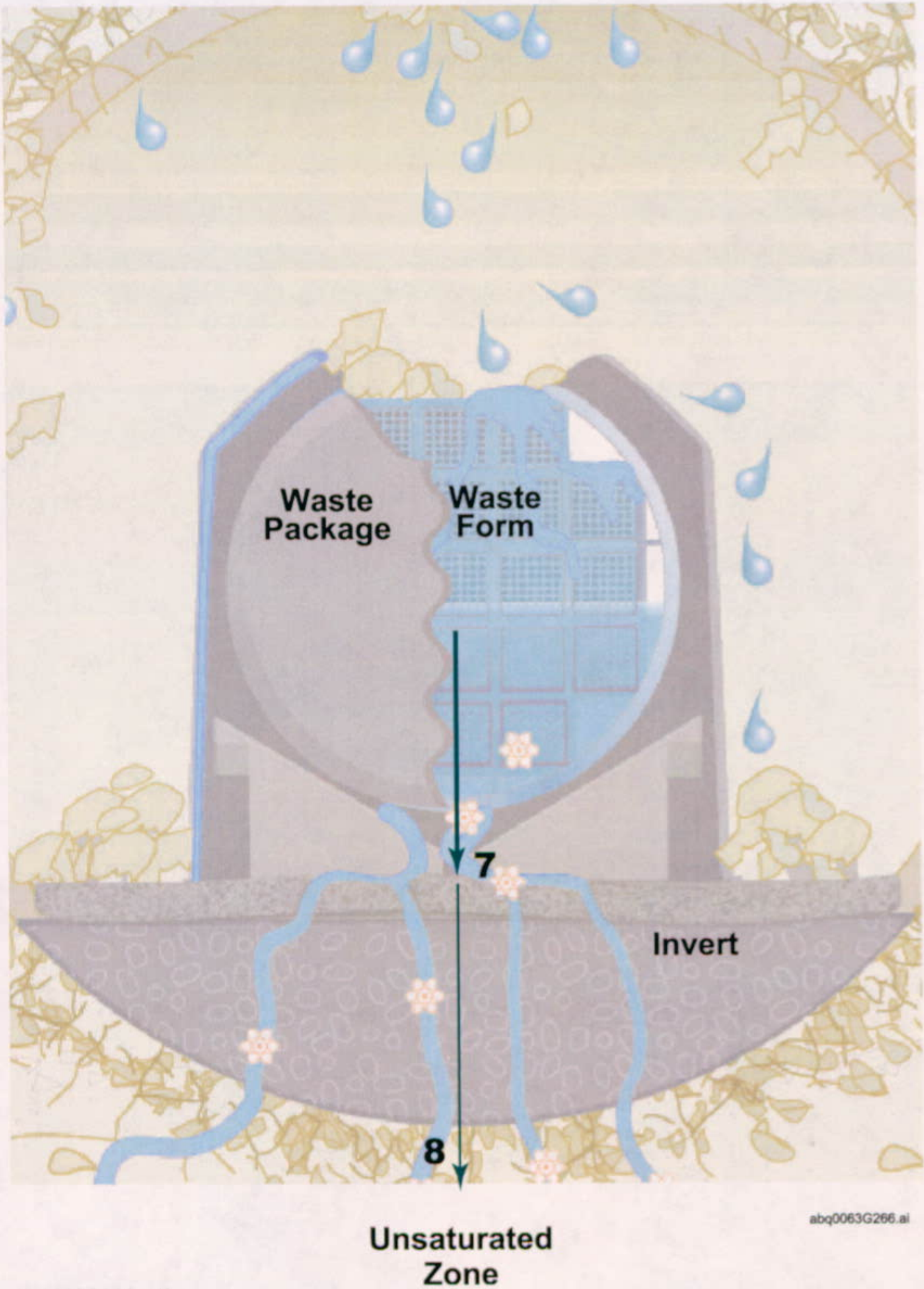

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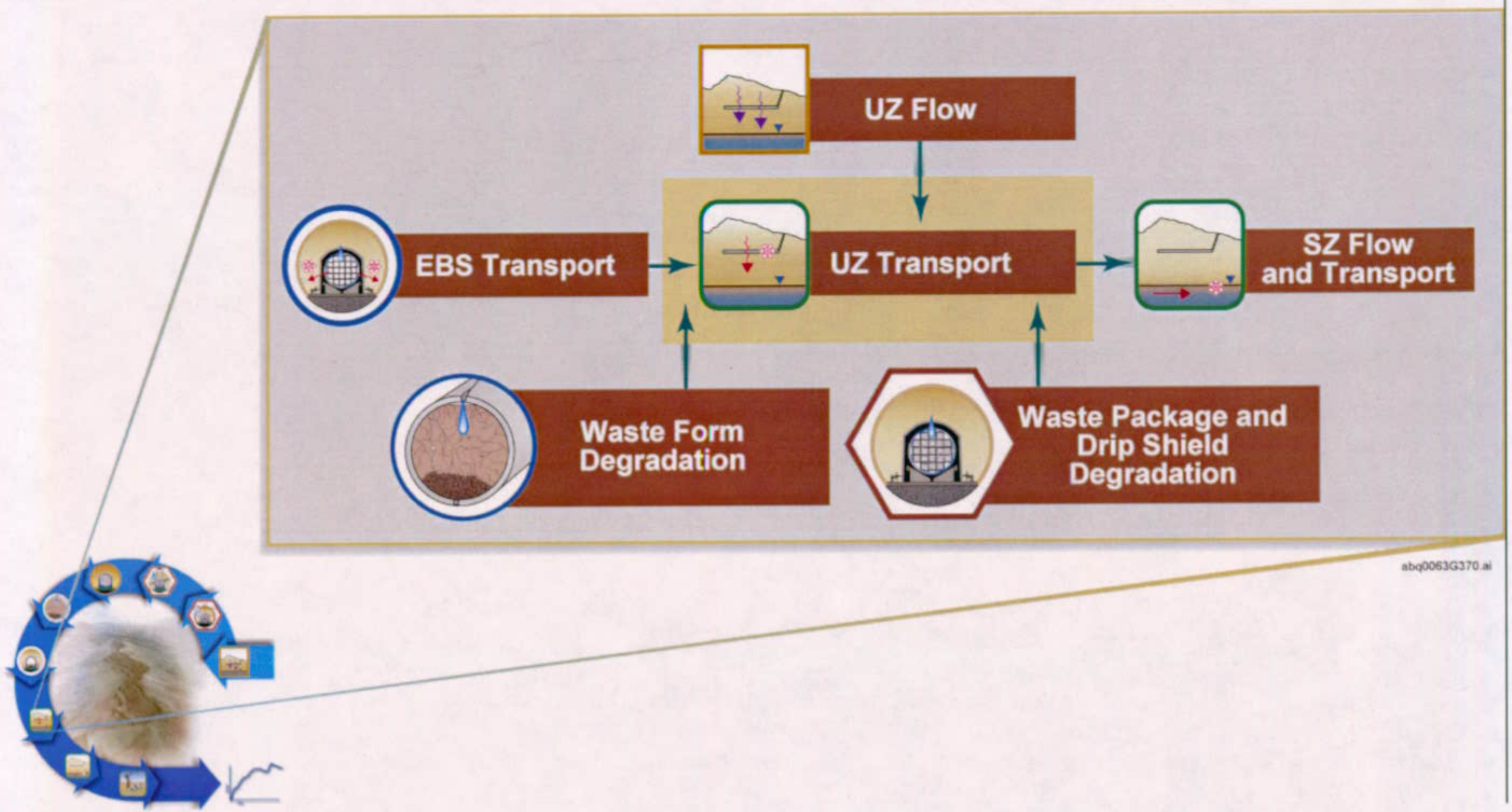
Figure 3.6-8. Schematic Diagram of the Transport Pathways in the Engineered Barrier System Transport Abstraction



abq0063G370

NOTE: Index figure in lower left is same as Figure 2.1-6

Figure 3.7-1. Information Flow Diagram for Unsaturated Zone Transport

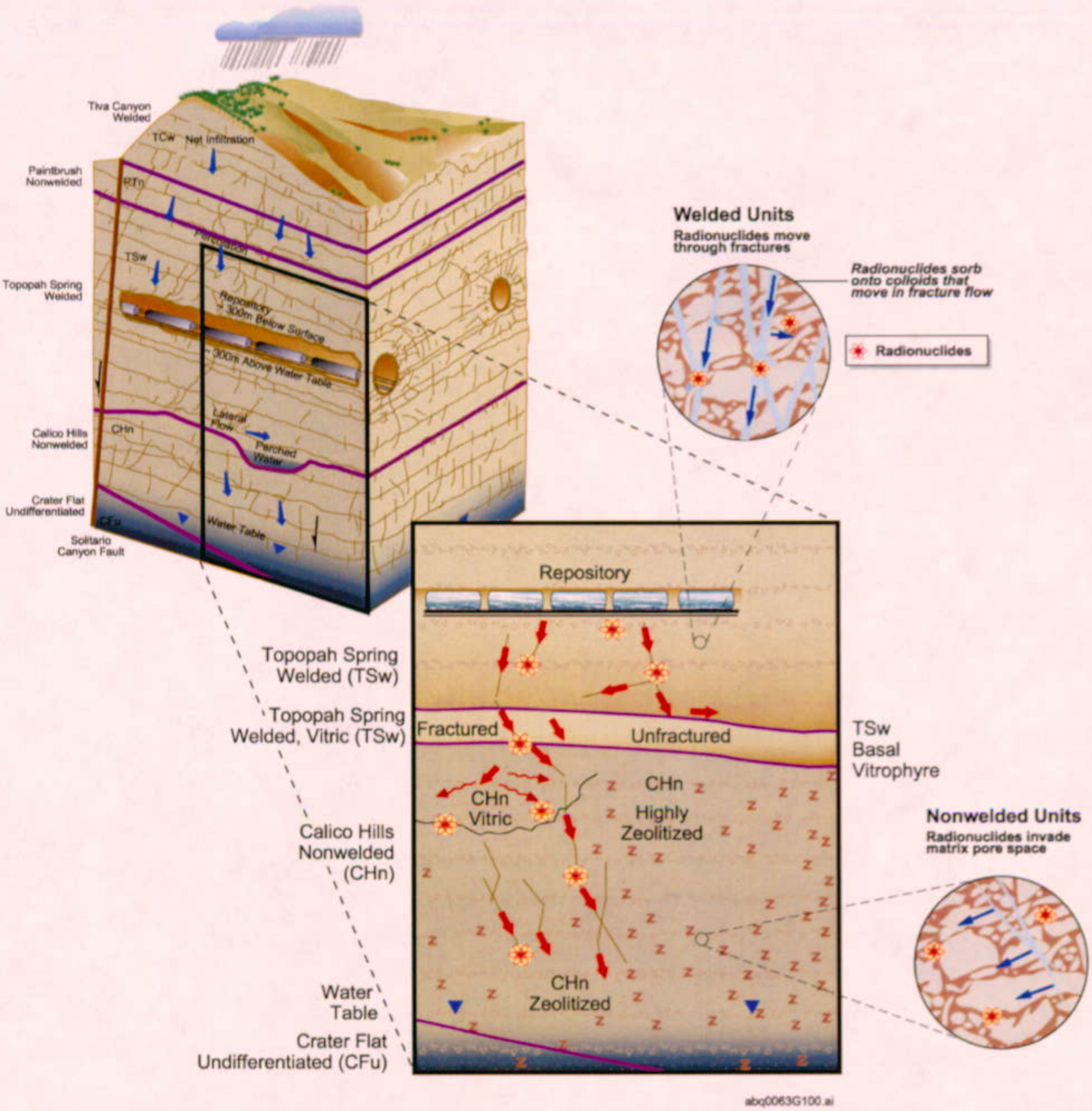


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Figure 3.7-2. Conceptual Drawing of Unsaturated Zone Transport Processes

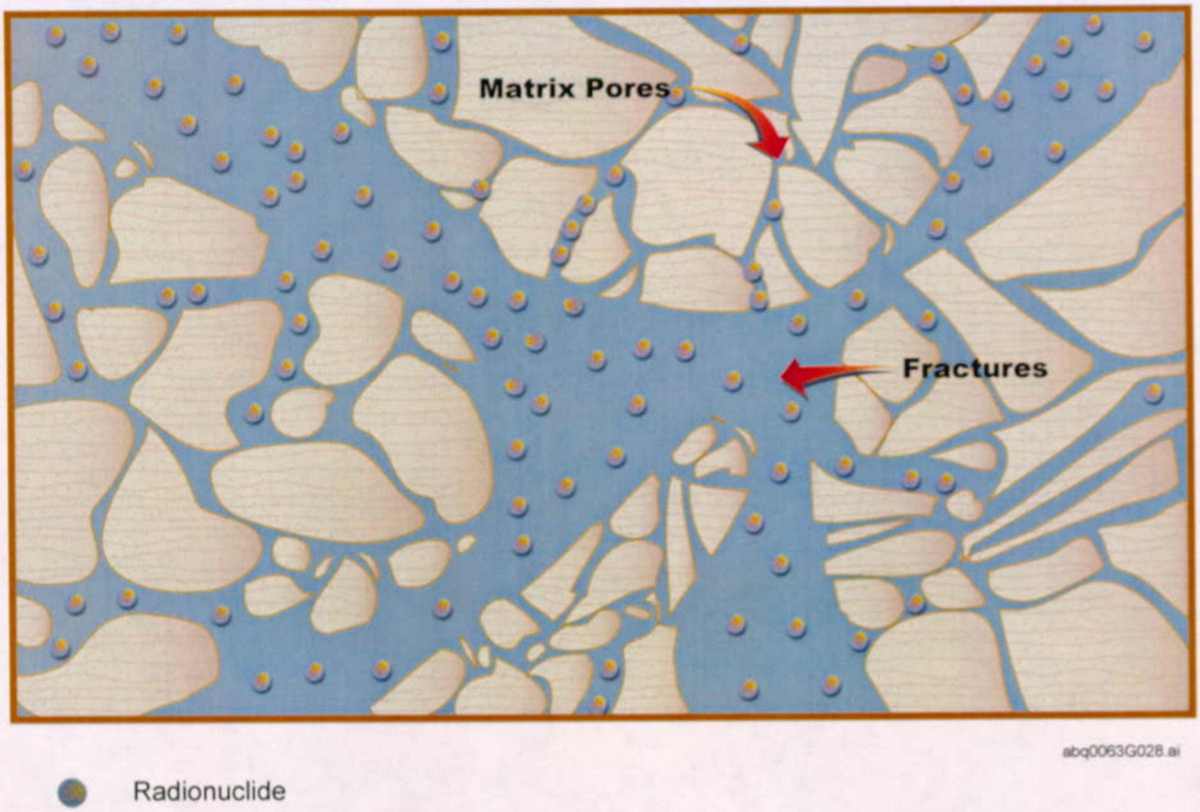

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Figure 3.7-3. Conceptual Drawing of Diffusion into Matrix Pores

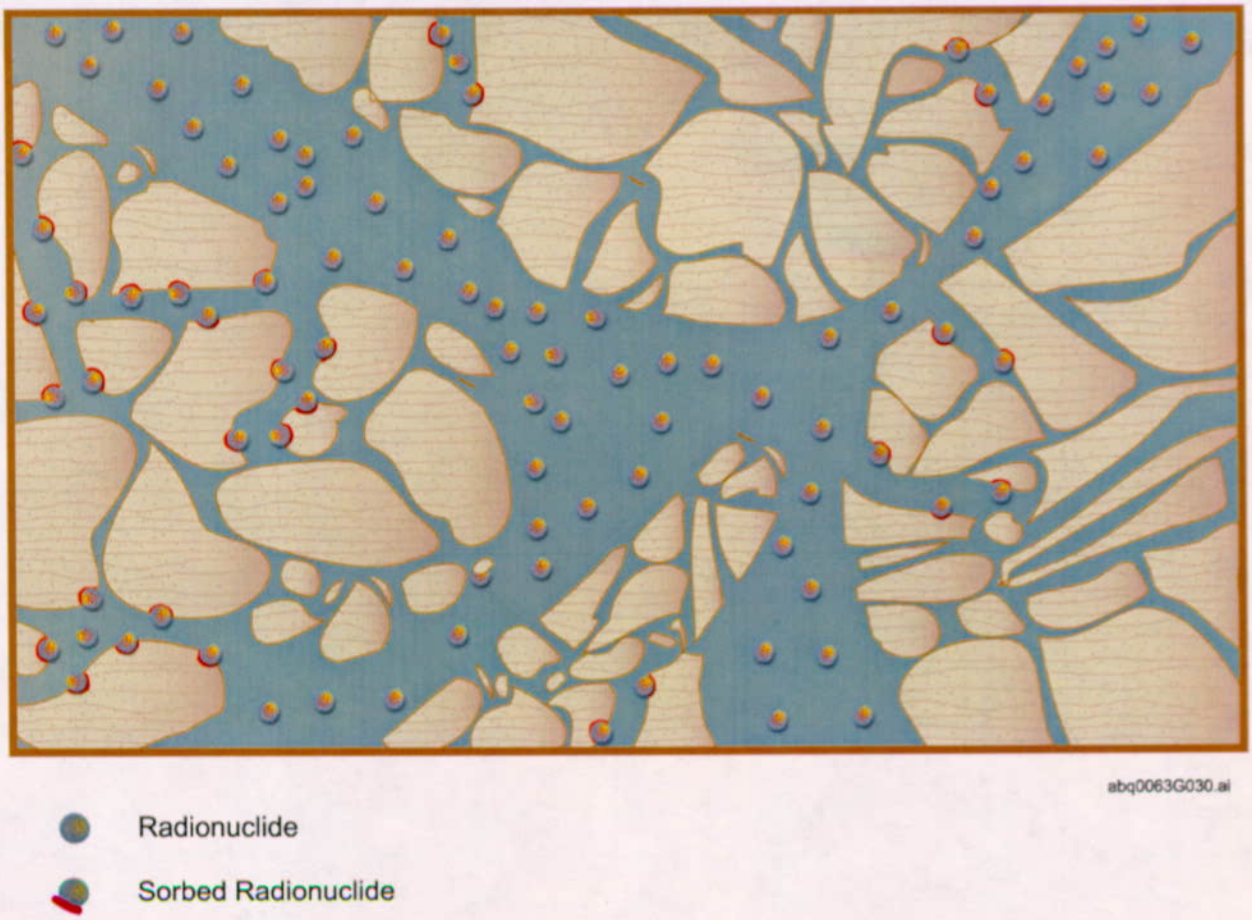

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Figure 3.7-4. Conceptual Drawing of Radionuclide Sorption

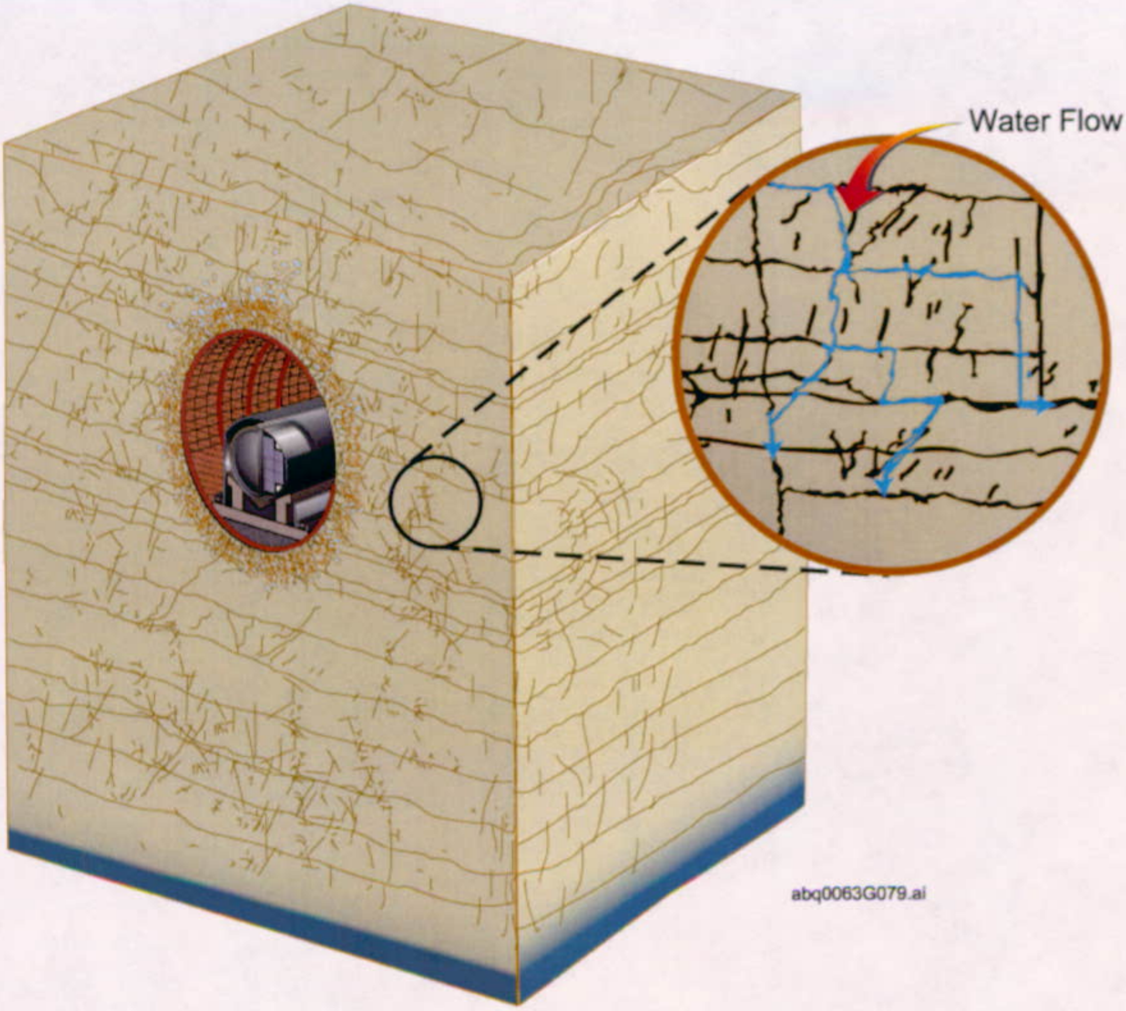

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Figure 3.7-5. Conceptual Drawing of Hydrodynamic Dispersion

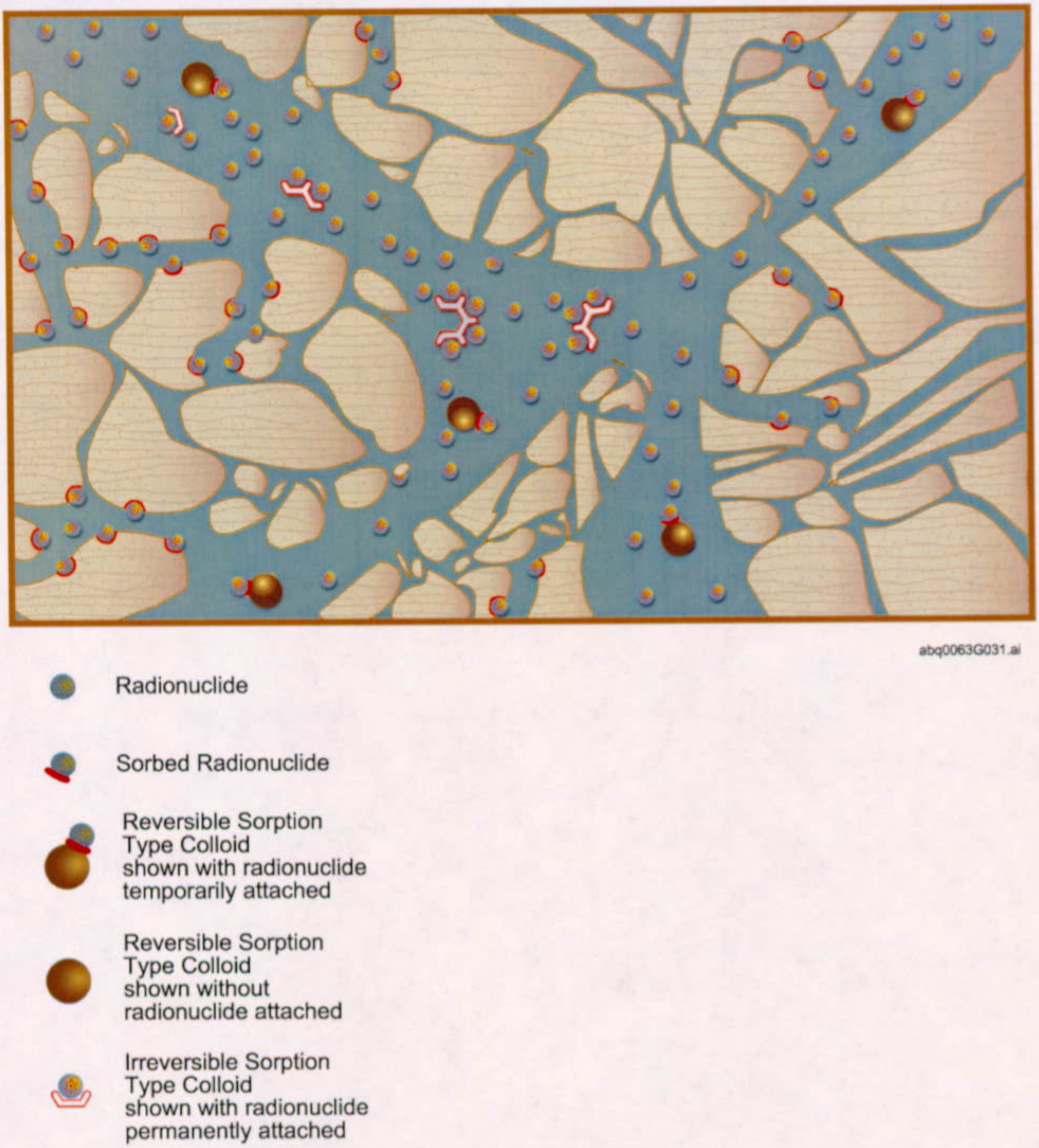

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Figure 3.7-6. Conceptual Drawing of Colloid-Facilitated Transport

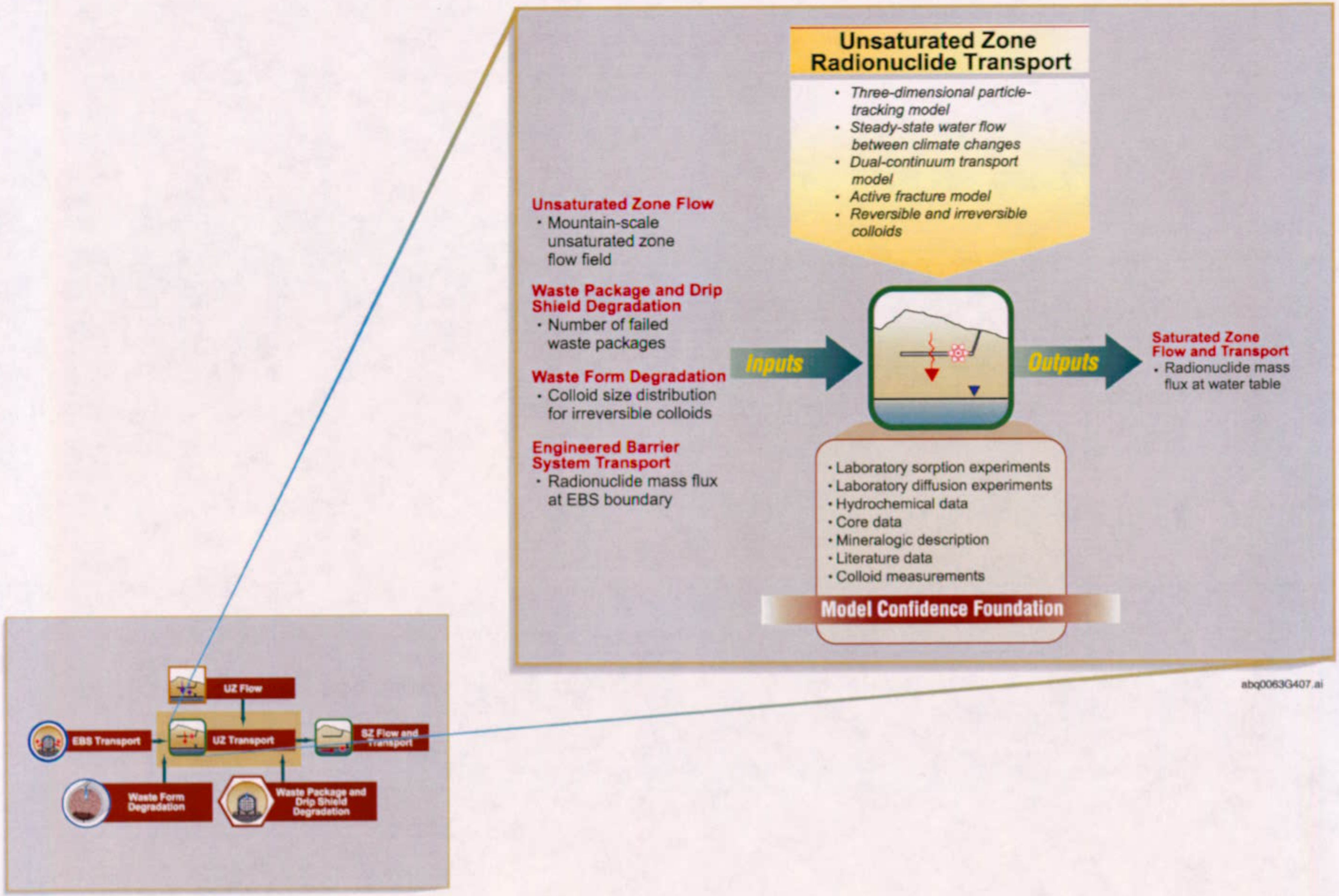

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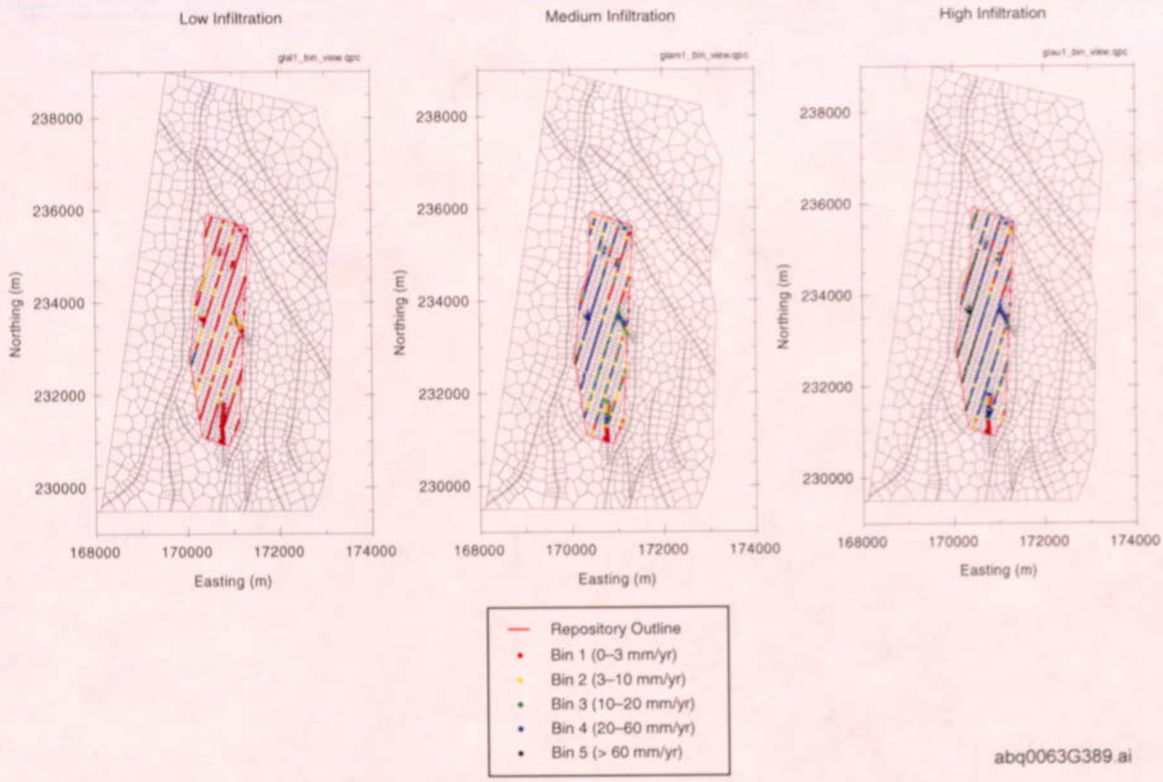
Figure 3.7-7. Connections between Unsaturated Zone Transport and Other Total System Performance Assessment Model Components



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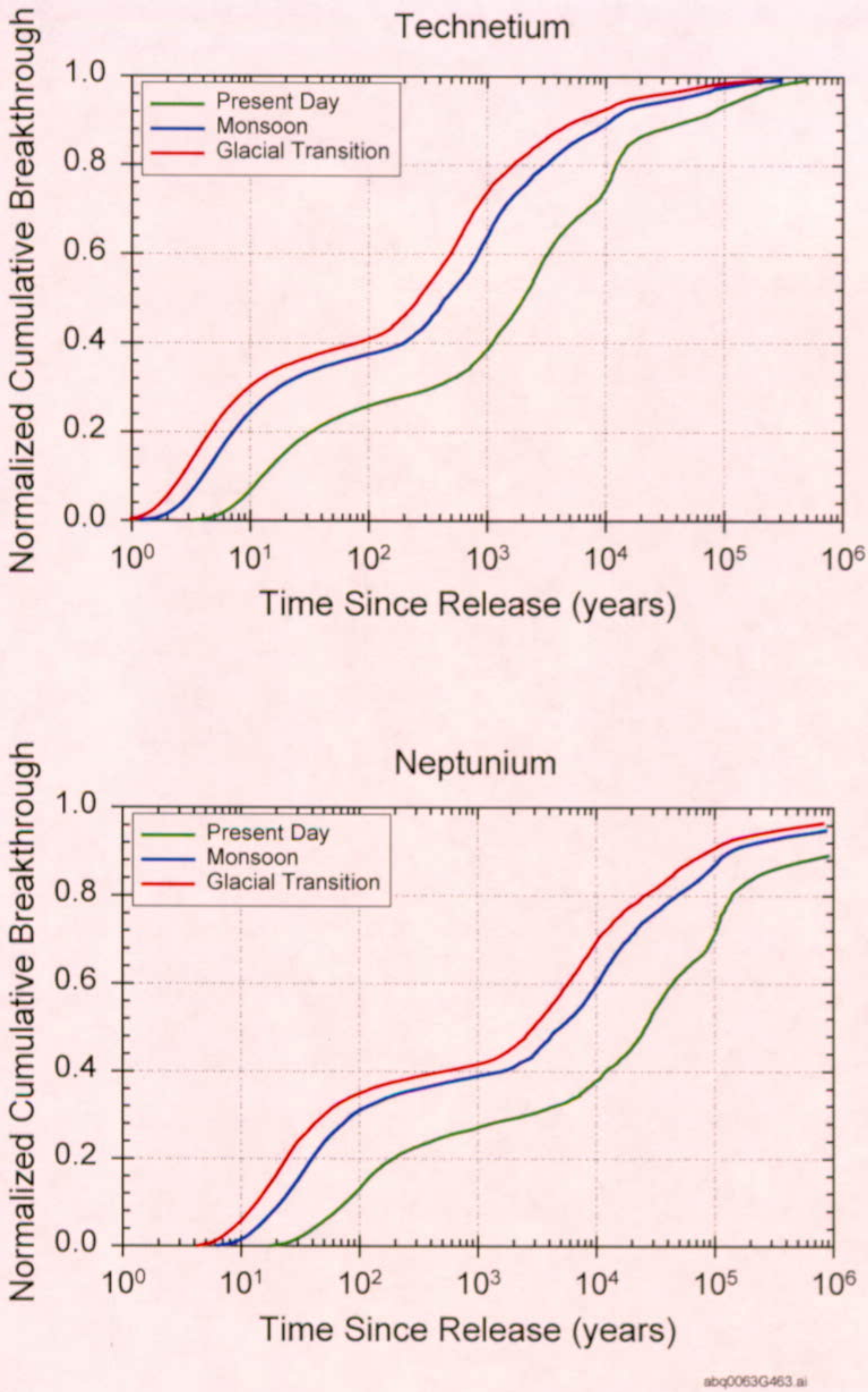
Source: Adapted from CRWMS M&O 2000 [148384], Figure 103

Figure 3.7-8. Radionuclide Release Locations for Five Infiltration Bins and Three Cases



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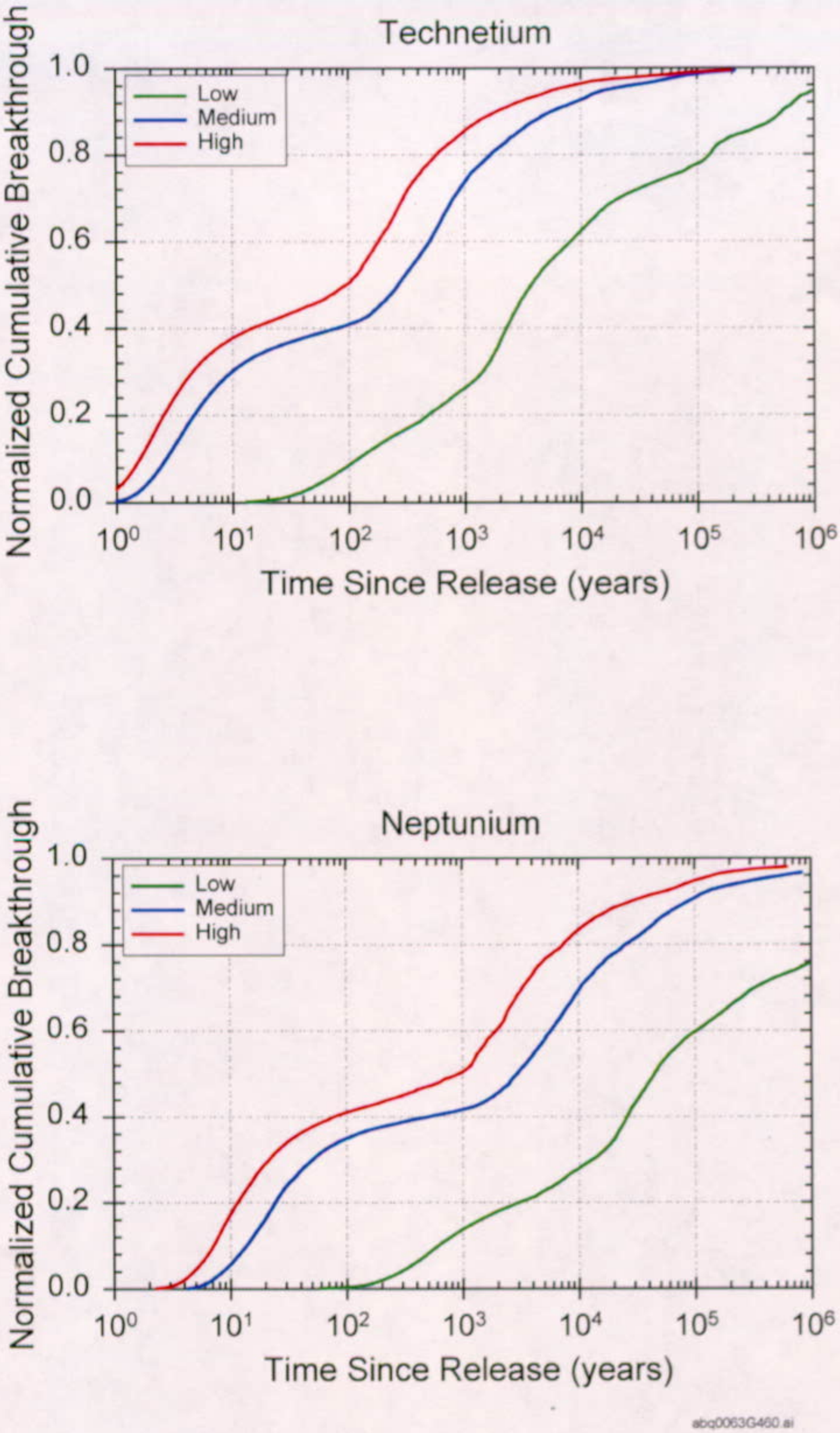
Source: Adapted from CRWMS M&O 2000 [134732], Figures 6 to 8

Figure 3.7-9. Breakthrough Curves for Three Climate States, Medium-Infiltration Case



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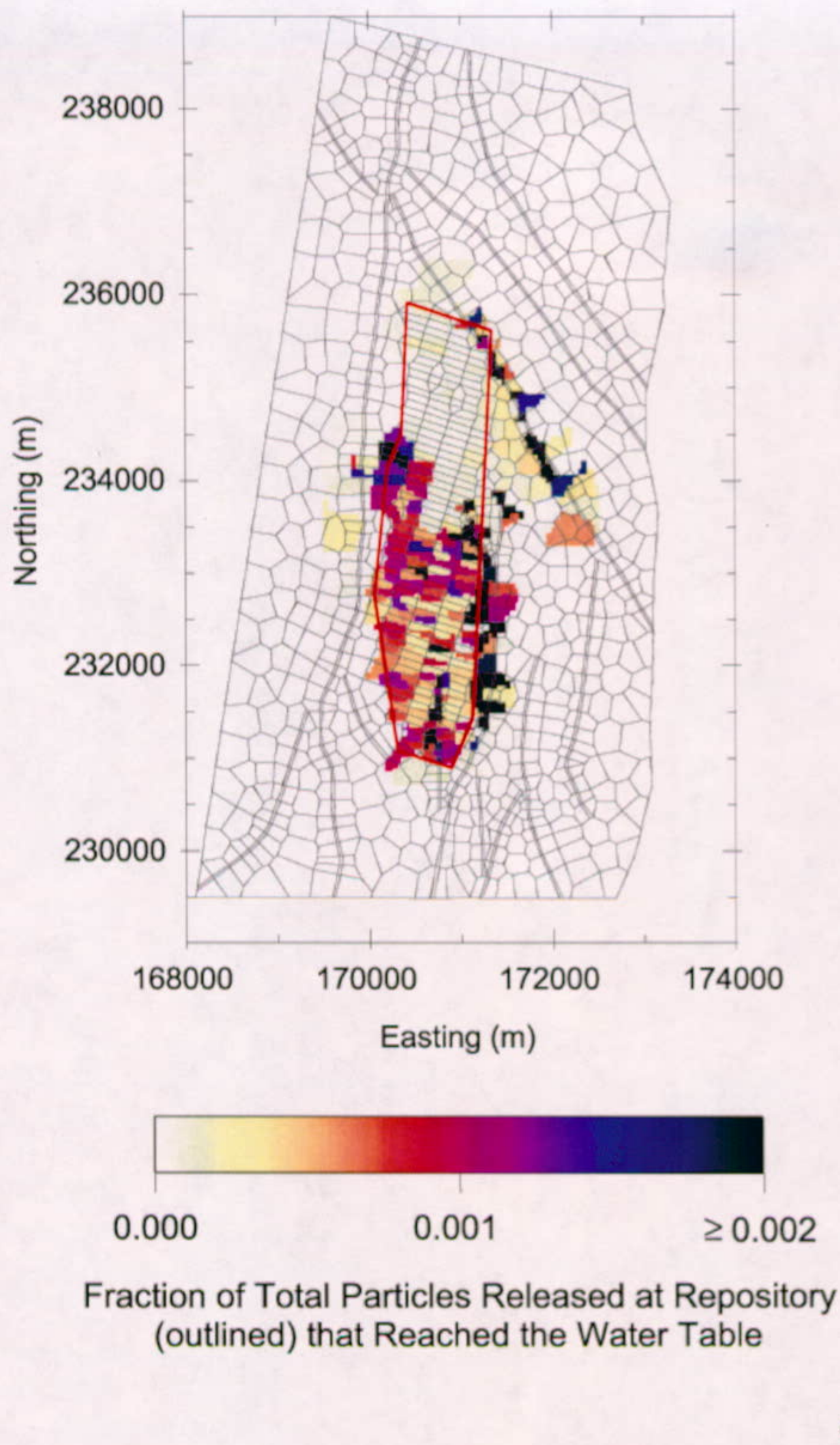
Source: Adapted from CRWMS M&O 2000 [134732], Figure 14

Figure 3.7-10. Breakthrough Curves for Three Infiltration Cases, Glacial-Transition Climate



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Source: Adapted from CRWMS M&O 2000 [134732], Figure 4

Figure 3.7-11. Locations of Particle Breakthrough at the Water Table, Medium-Infiltration Case and Glacial-Transition Climate

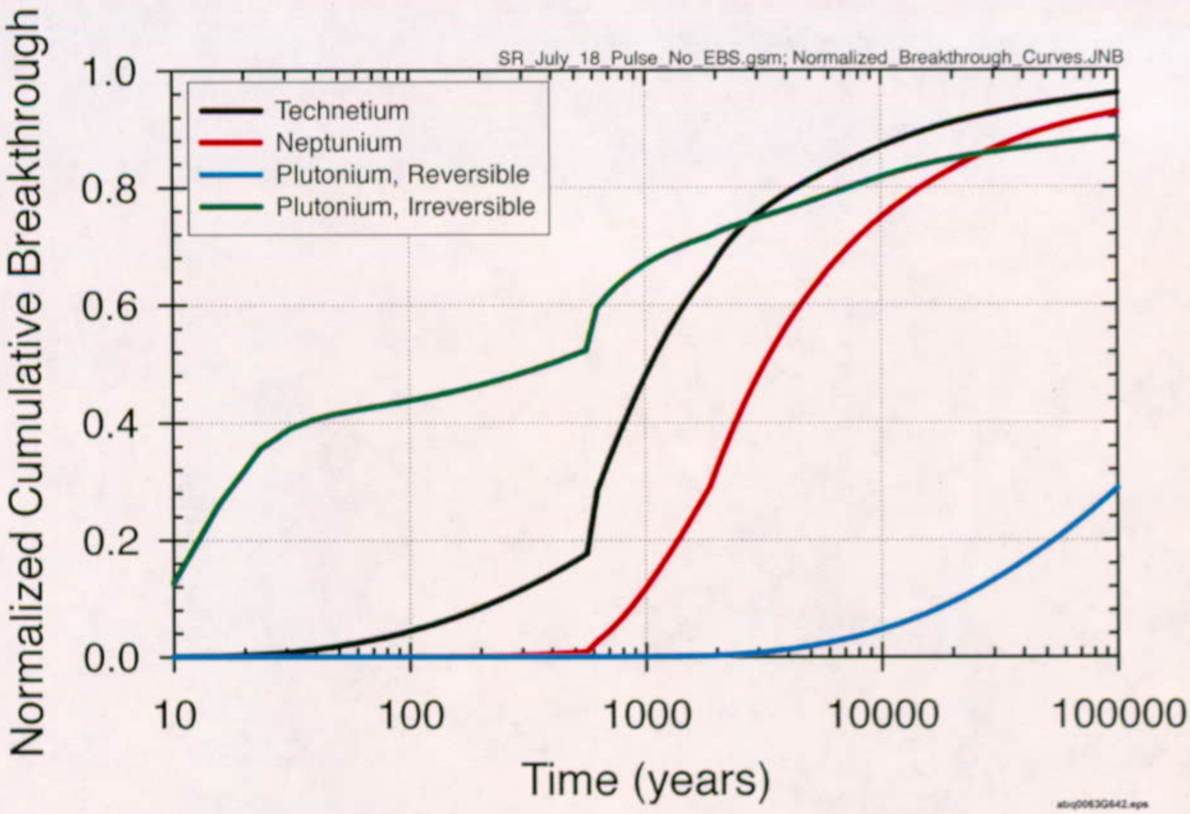

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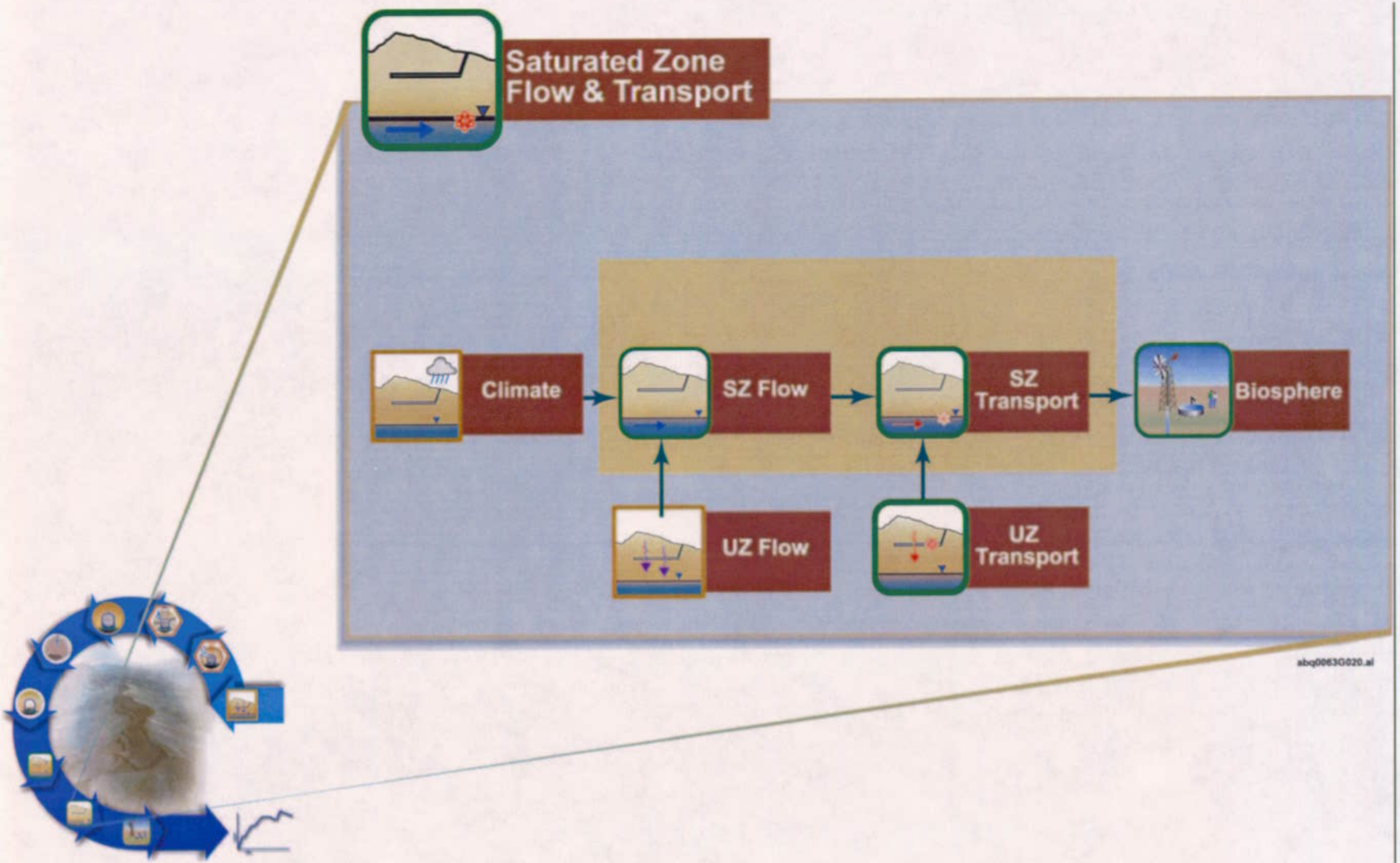
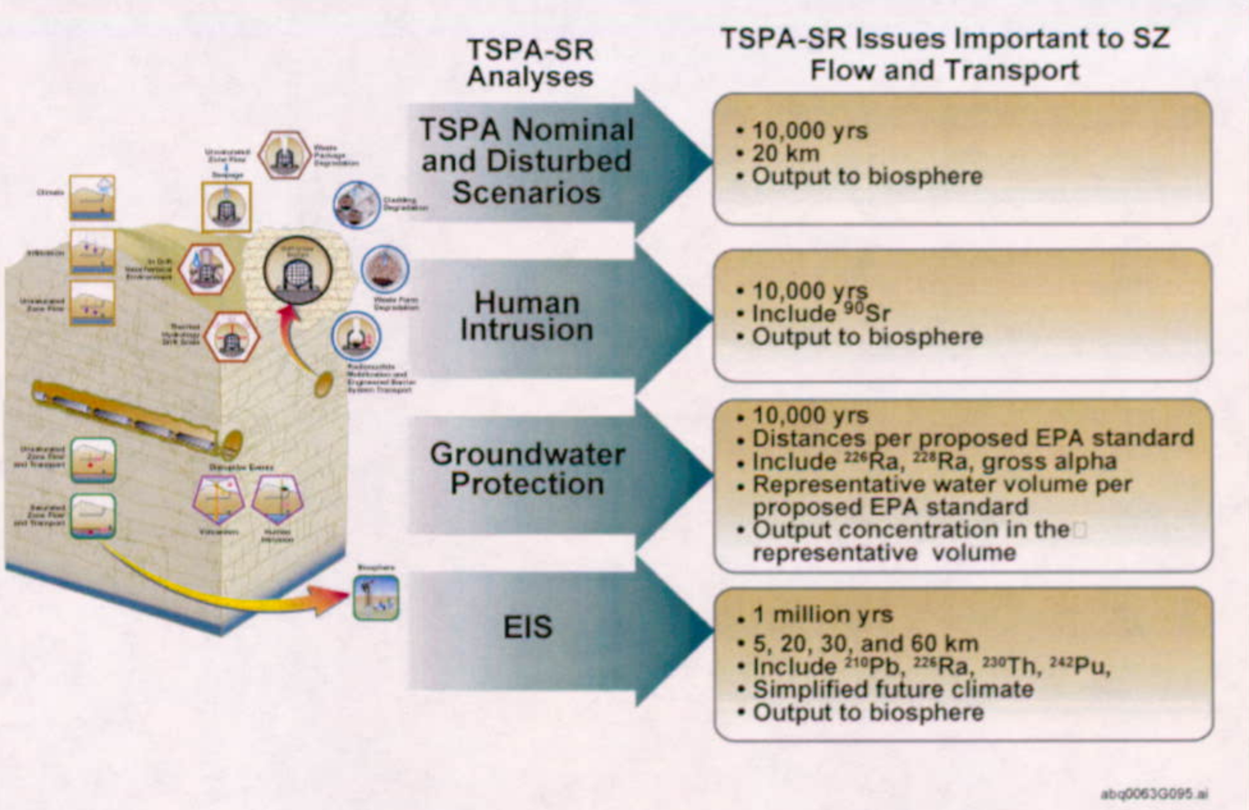
Figure 3.7-12. Mean Breakthrough Curves for 100 Realizations of Unsaturated Zone Transport



abq0063G020

NOTE: Index figure in lower left is same as Figure 2.1-6

Figure 3.8-1. Diagram of the Saturated Zone Component and Its Relationship with Other Total System Performance Assessment Components



abq0063G095

Figure 3.8-2. Saturated Zone-Component Emphasis is Different for the Four Major Site Recommendation Analyses

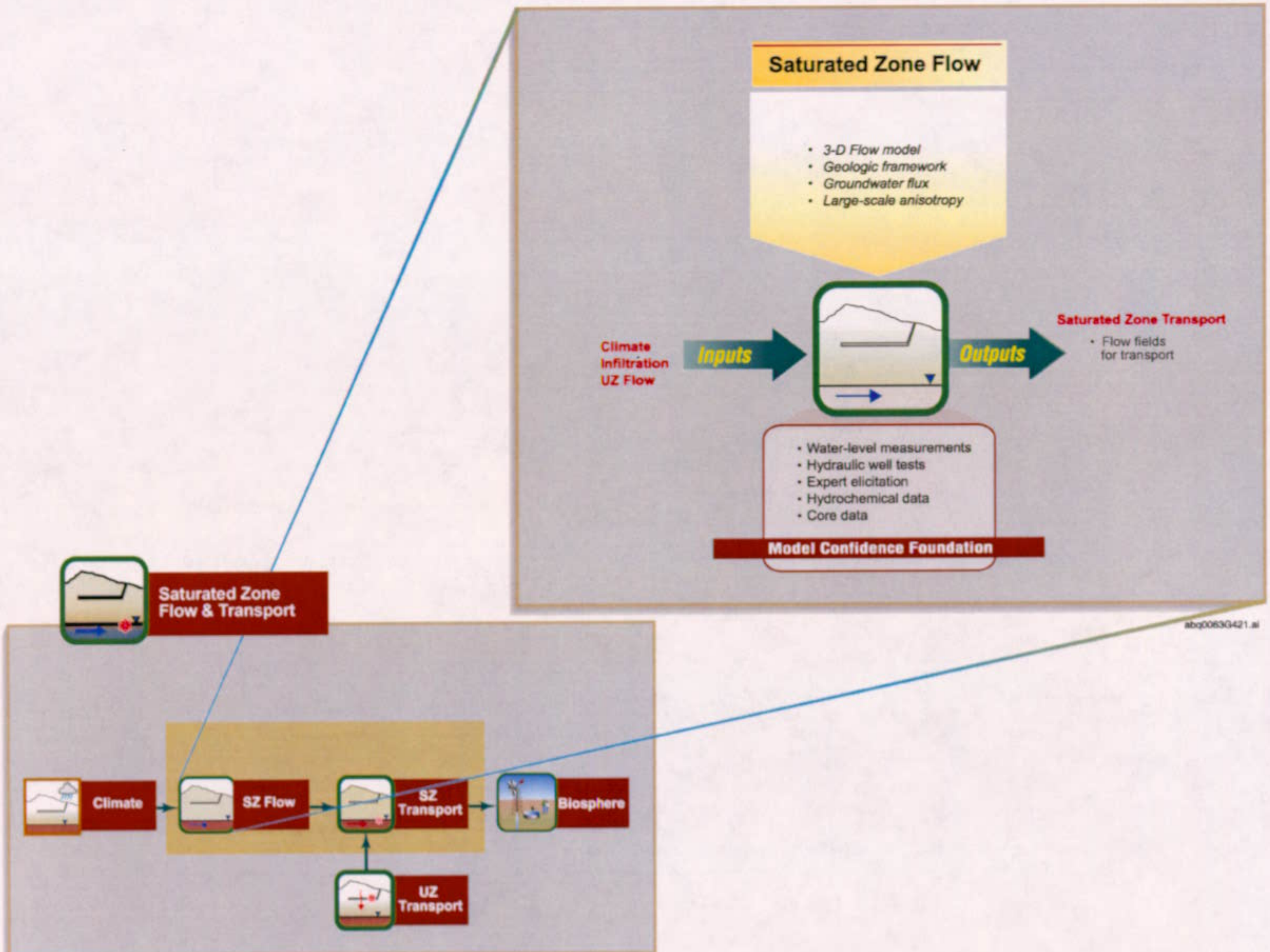


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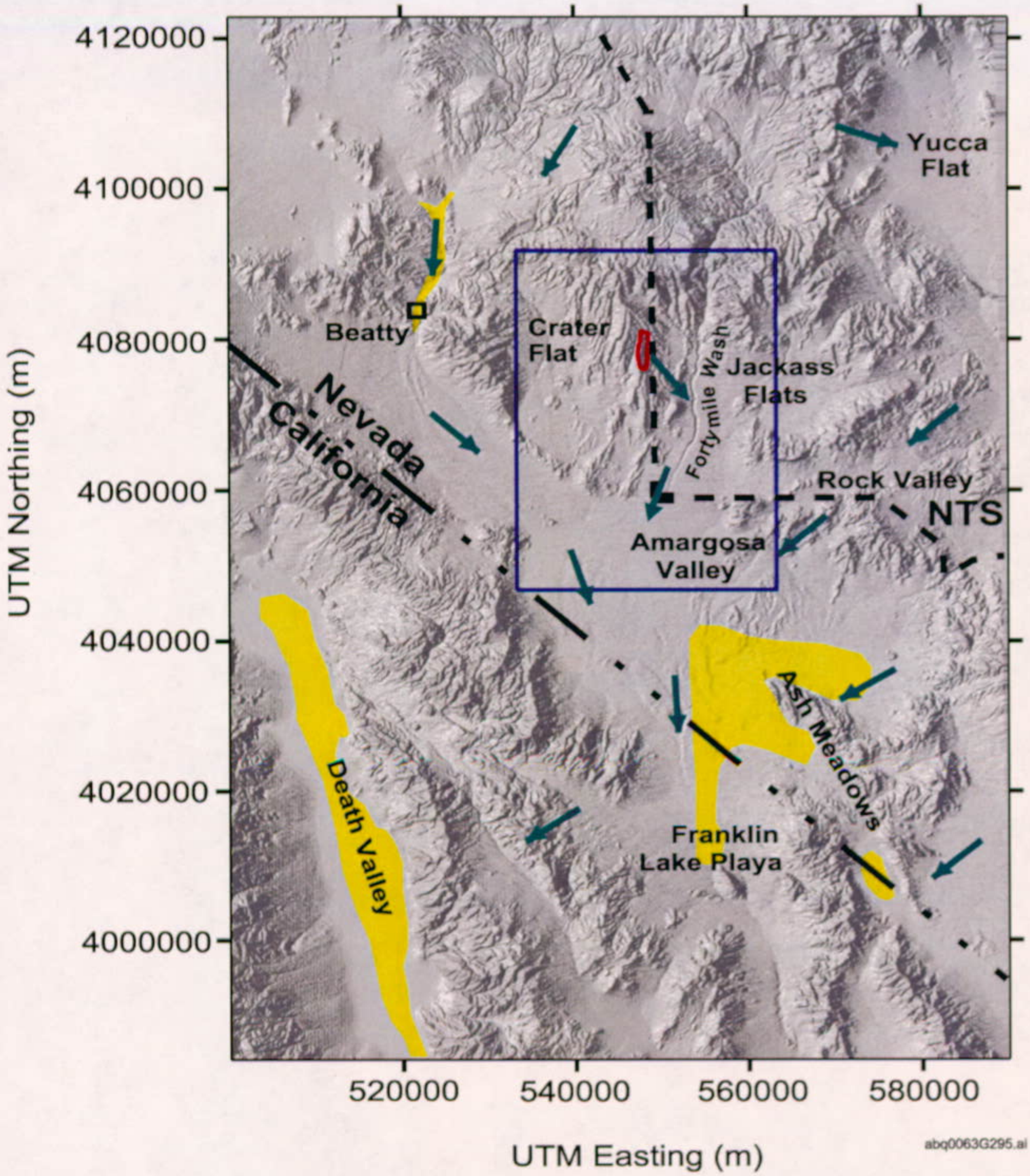
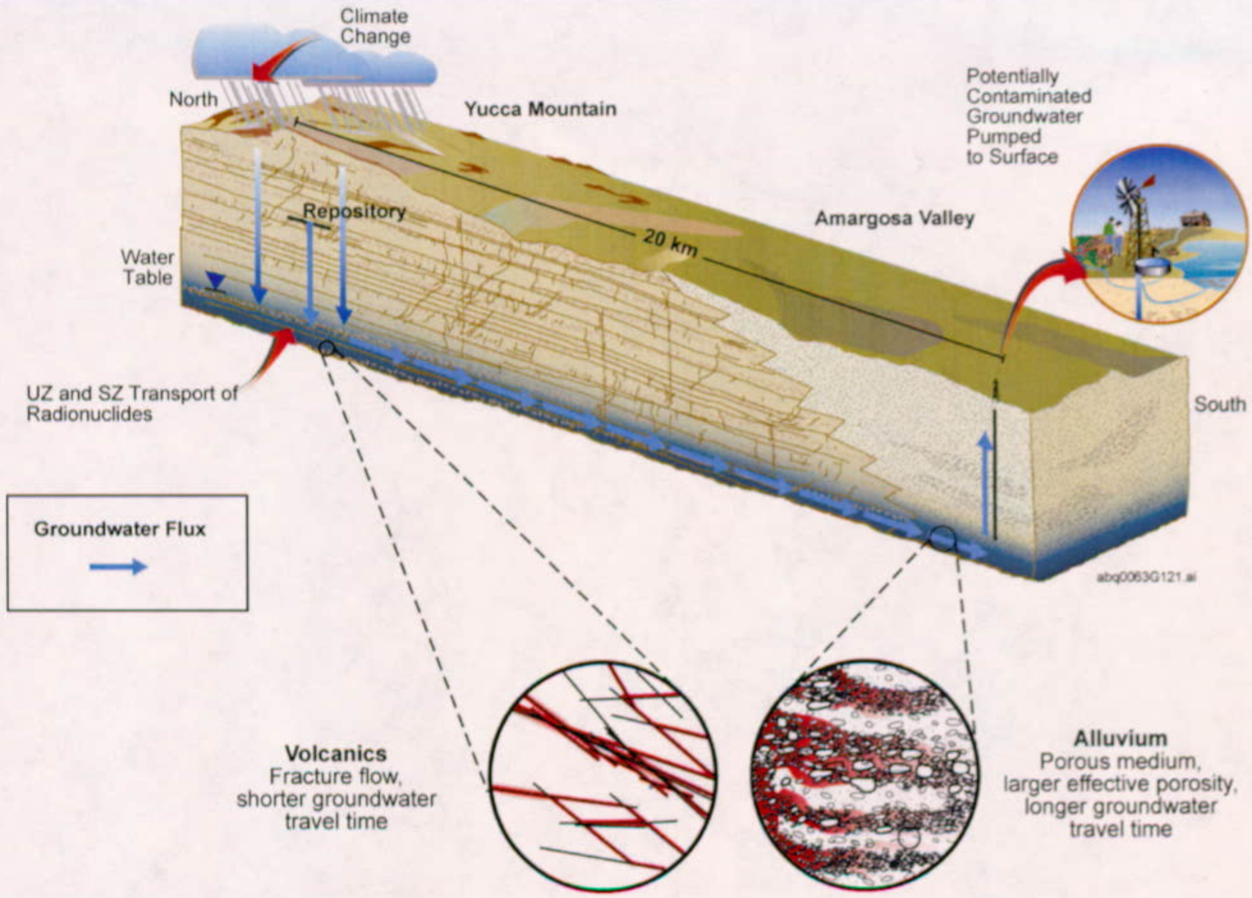
Figure 3.8-3. Summary of Inputs and Outputs for the Saturated Zone Flow Component



abq0063G295

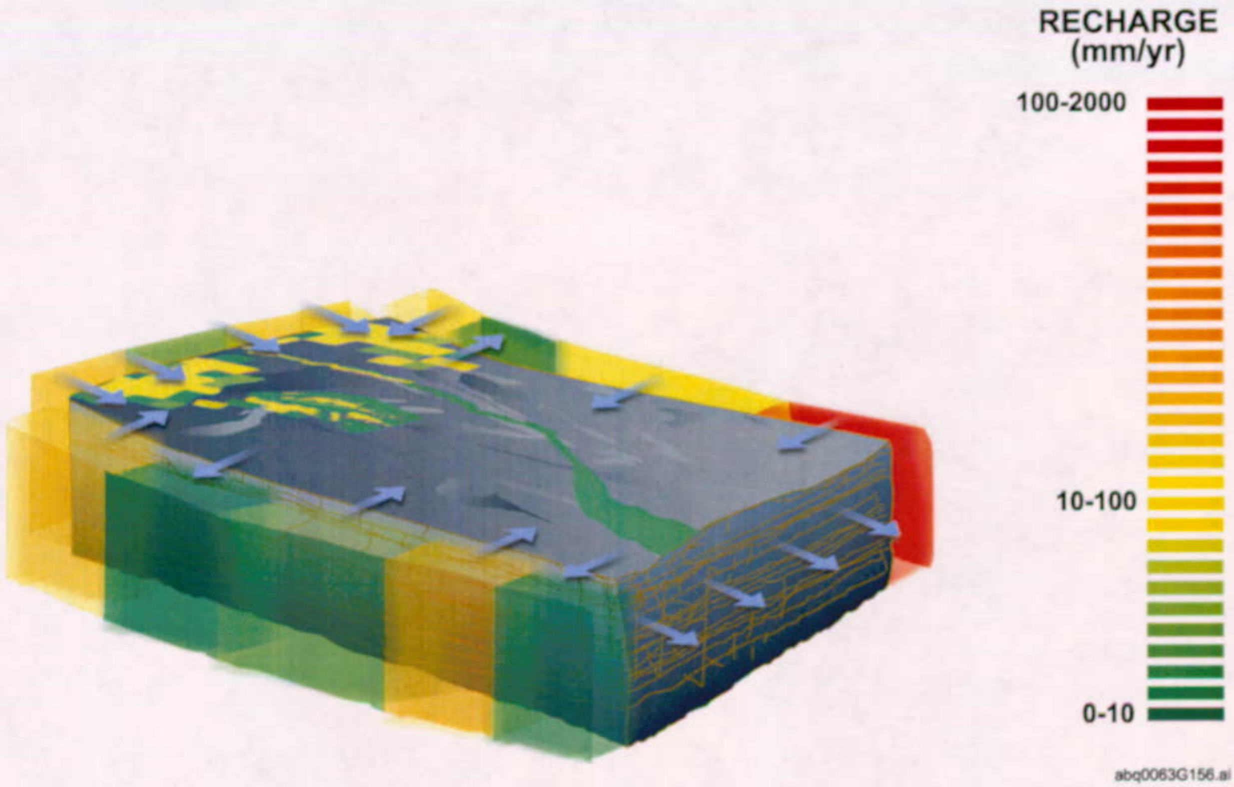
NOTE: The purple line represents the model domain; the green arrows show the general direction of groundwater flow; the yellow areas indicate regions of discharge; and the red area indicates the location of the potential repository.

Figure 3.8-4. Regional Map of the Saturated Zone Flow System Showing Direction of Flow and Outline of the Three-Dimensional Saturated Zone Flow Model Domain



abq0063G121

Figure 3.8-5. Conceptualization of Saturated Zone Flow



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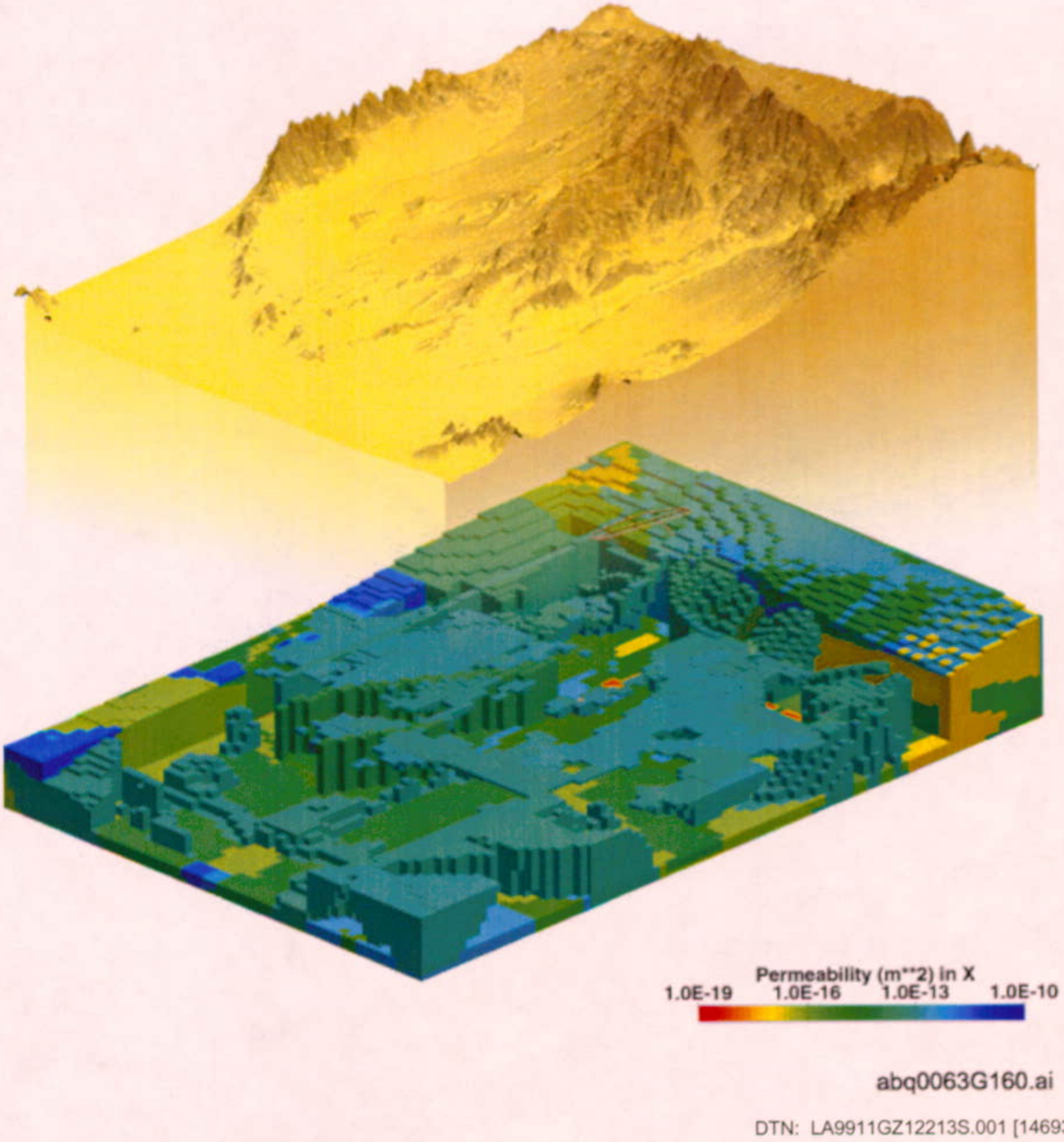
DTN: SN9908T0581999.001 [132867]

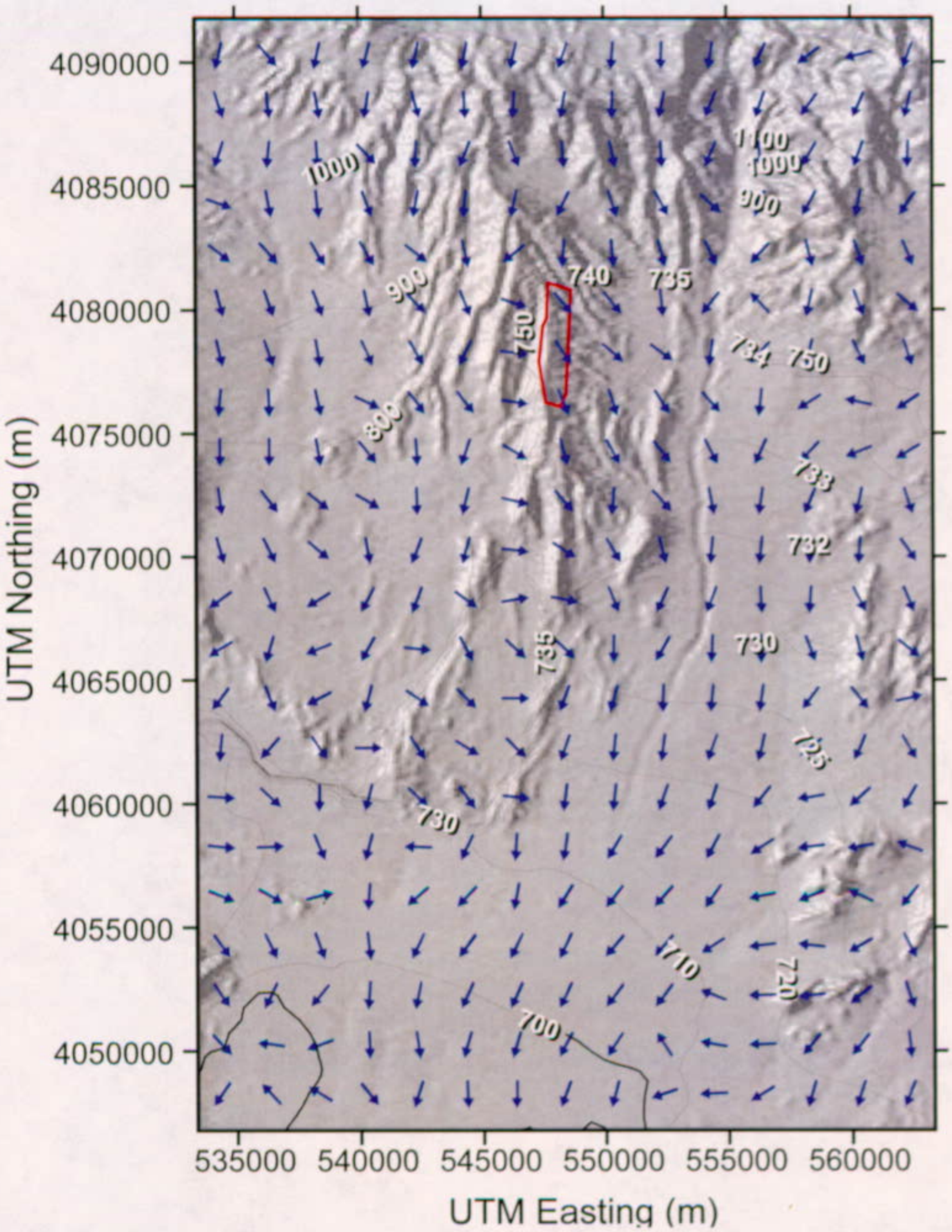
Figure 3.8-6. Lateral and Top Boundary Conditions for the Three-Dimensional Saturated Zone Flow Model for the Present-Day Climate



abq0063G160

NOTE: Land surface is for illustrative purposes only.

Figure 3.8-7. Three-Dimensional Saturated Zone Model Domain Showing the Different Permeability Fields



DTN: SN0004T0501600.005 [151515]

abq0063G304

Figure 3.8-8. Potentiometric Surface and Specific-Discharge Vectors Calculated by the Three-Dimensional Saturated Zone Flow Model for the Present-Day Climate

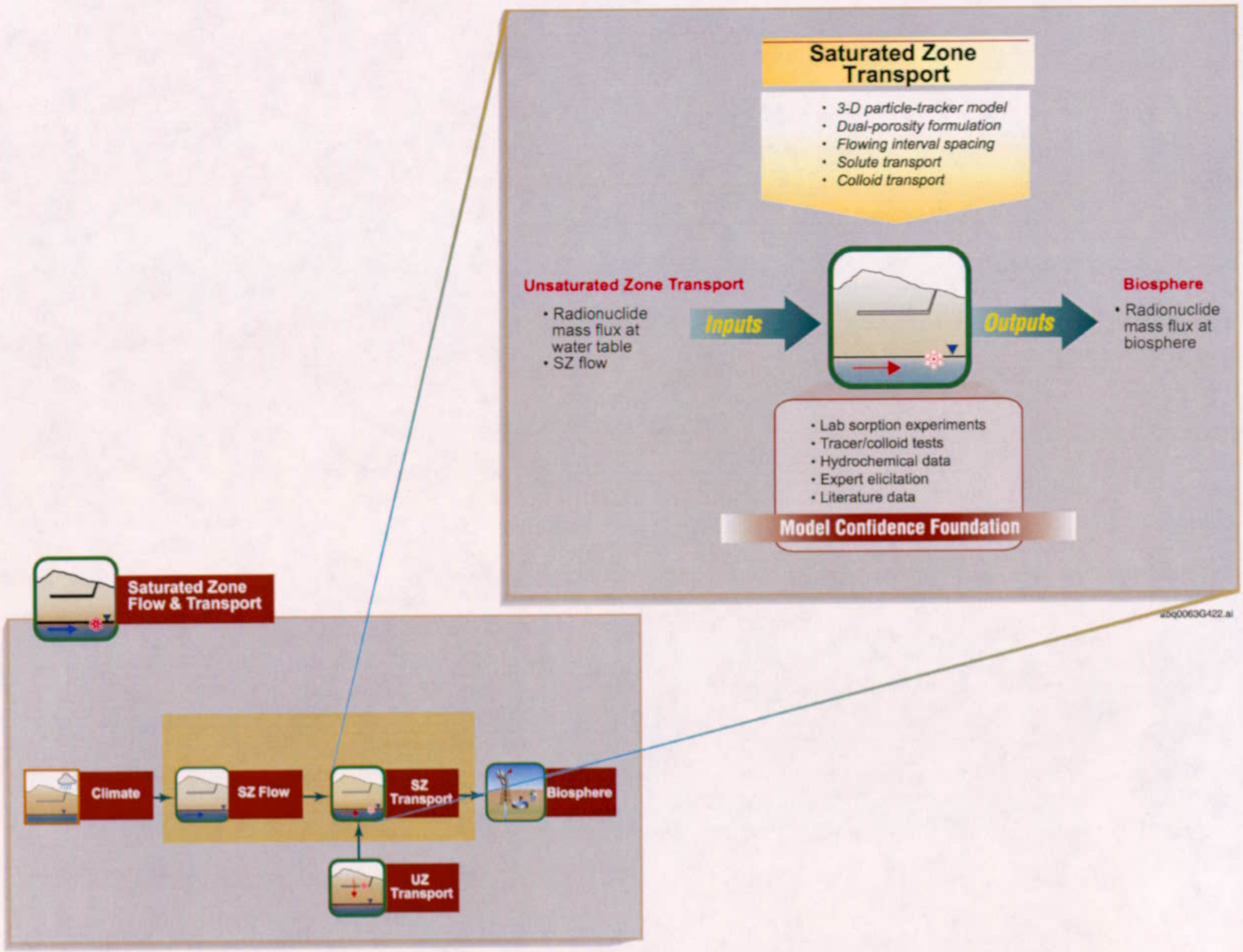


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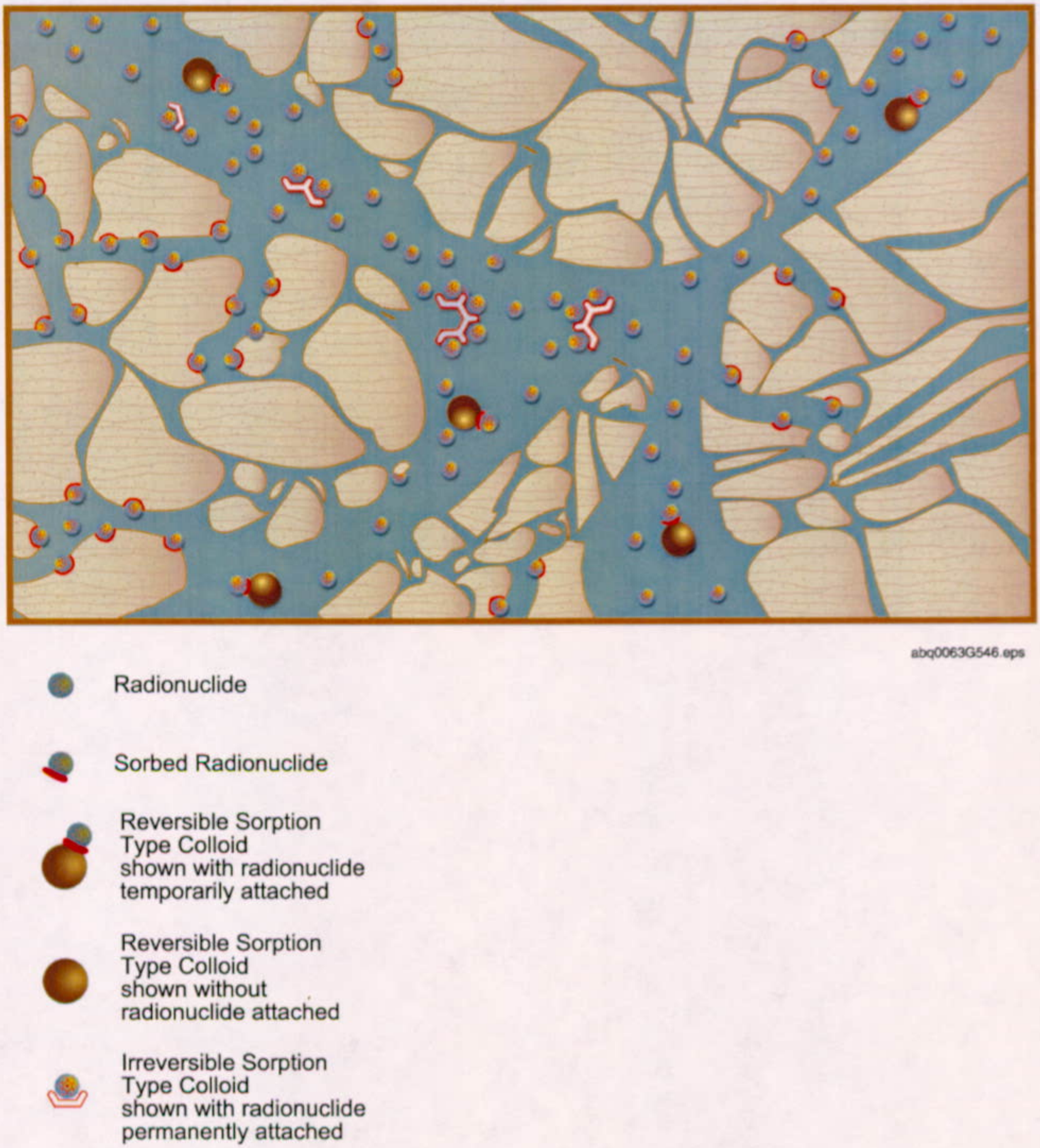
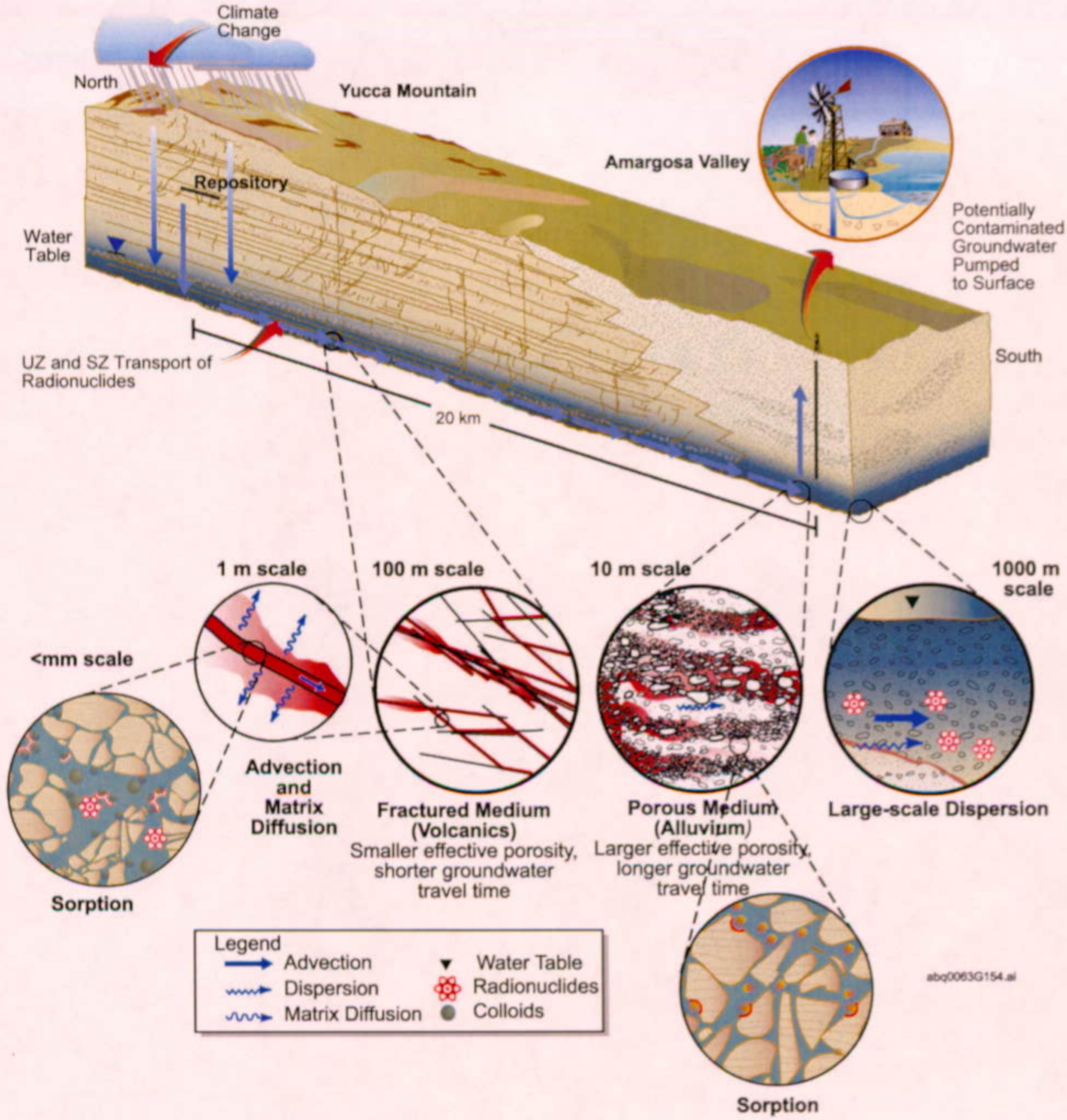
Figure 3.8-9. Summary of Inputs and Outputs for the Saturated Zone Transport Component



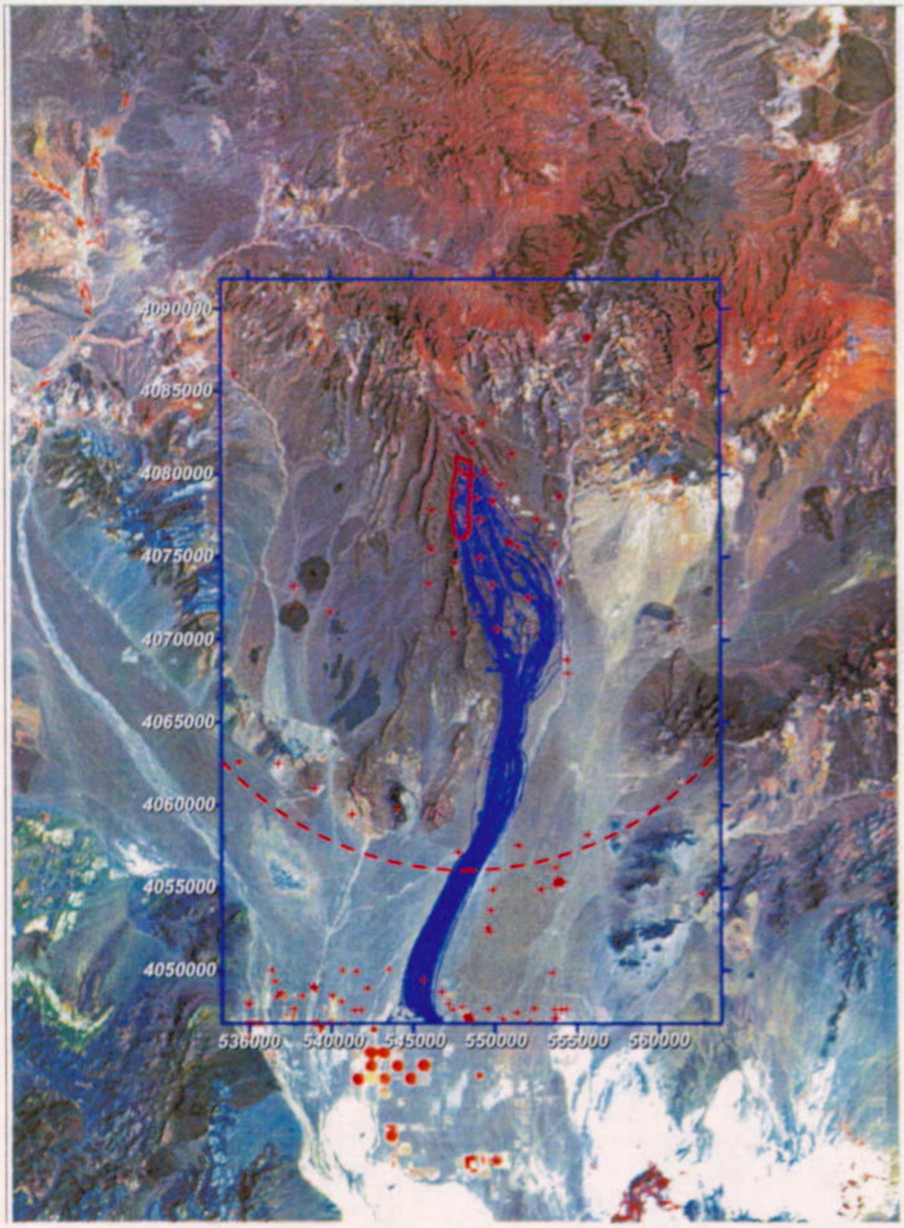
abq0063G546

Figure 3.8-10. Illustration of Colloid Facilitated Transport Mechanisms



abq0063G154

Figure 3.8-11. Conceptualization of Features and Processes Important to Saturated Zone Transport

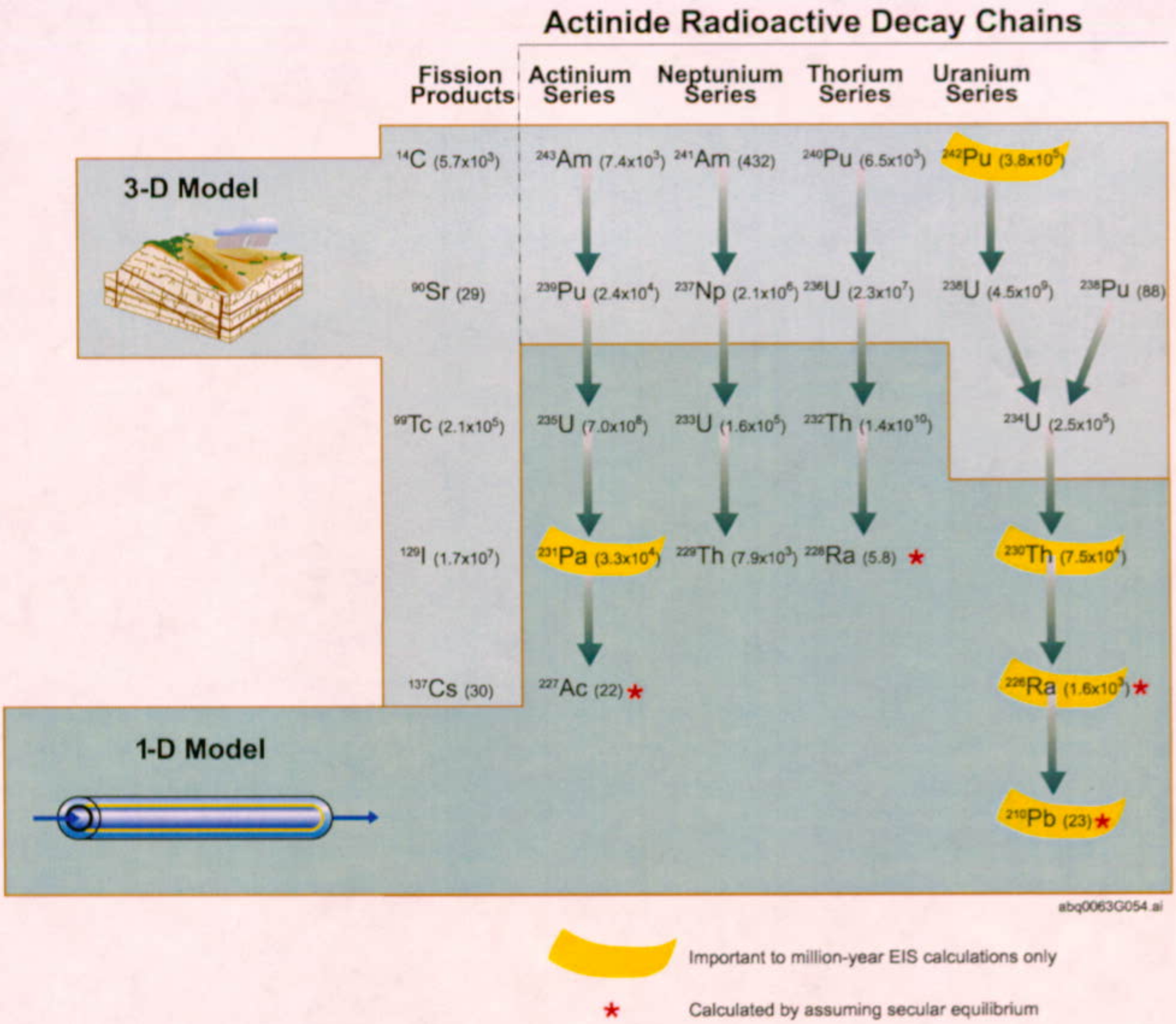


abq0063G155

DTN: SN0004T0501600.005 [151515]

NOTE: The red dashed line represents the 20-km boundary; the solid red line is the potential repository outline; the red crosses indicate borehole locations; and the blue rectangle outlines the three-dimensional saturated-zone model domain.

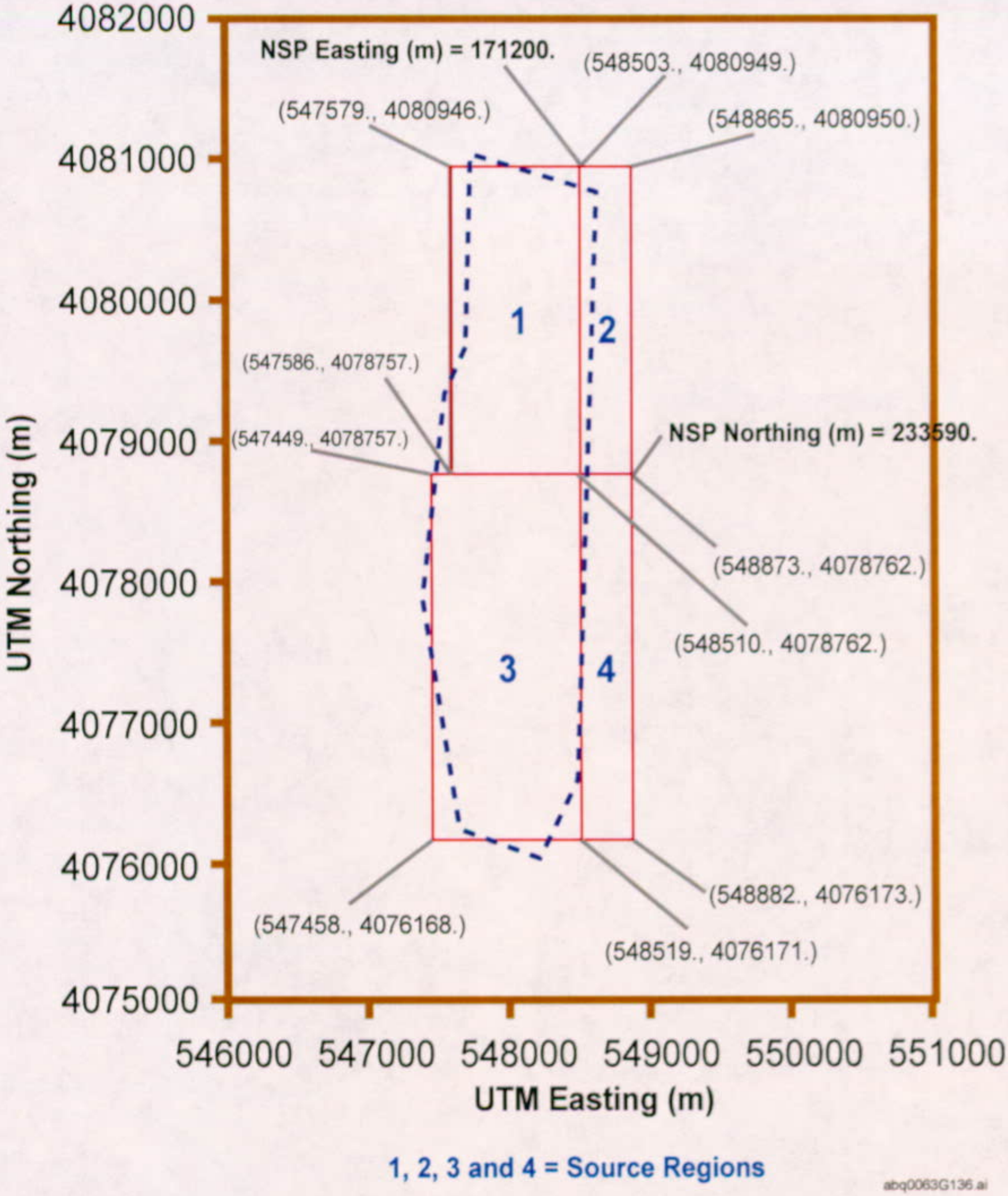
Figure 3.8-12. Map of the Three-Dimensional Saturated Zone Model Domain Showing Simulated Transport Particle Paths



abq0063G054

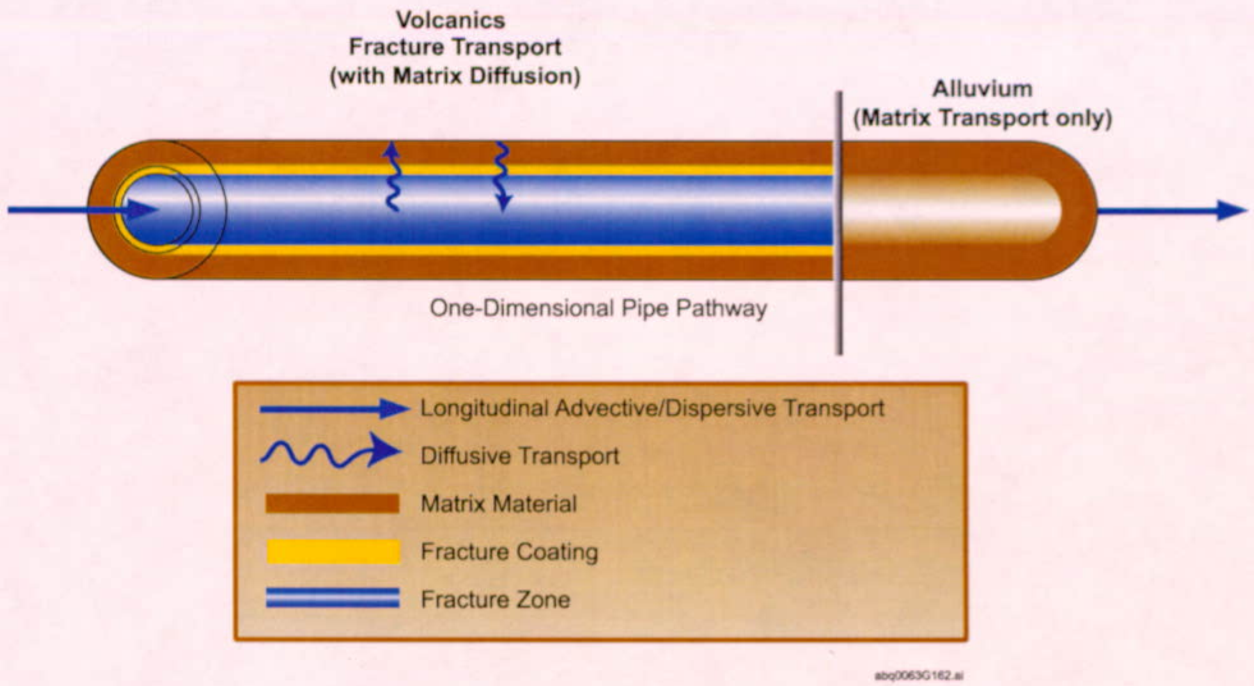
NOTE: Half-lives in years in parentheses.

Figure 3.8-13. Radionuclides Considered in Saturated Zone Transport Calculations



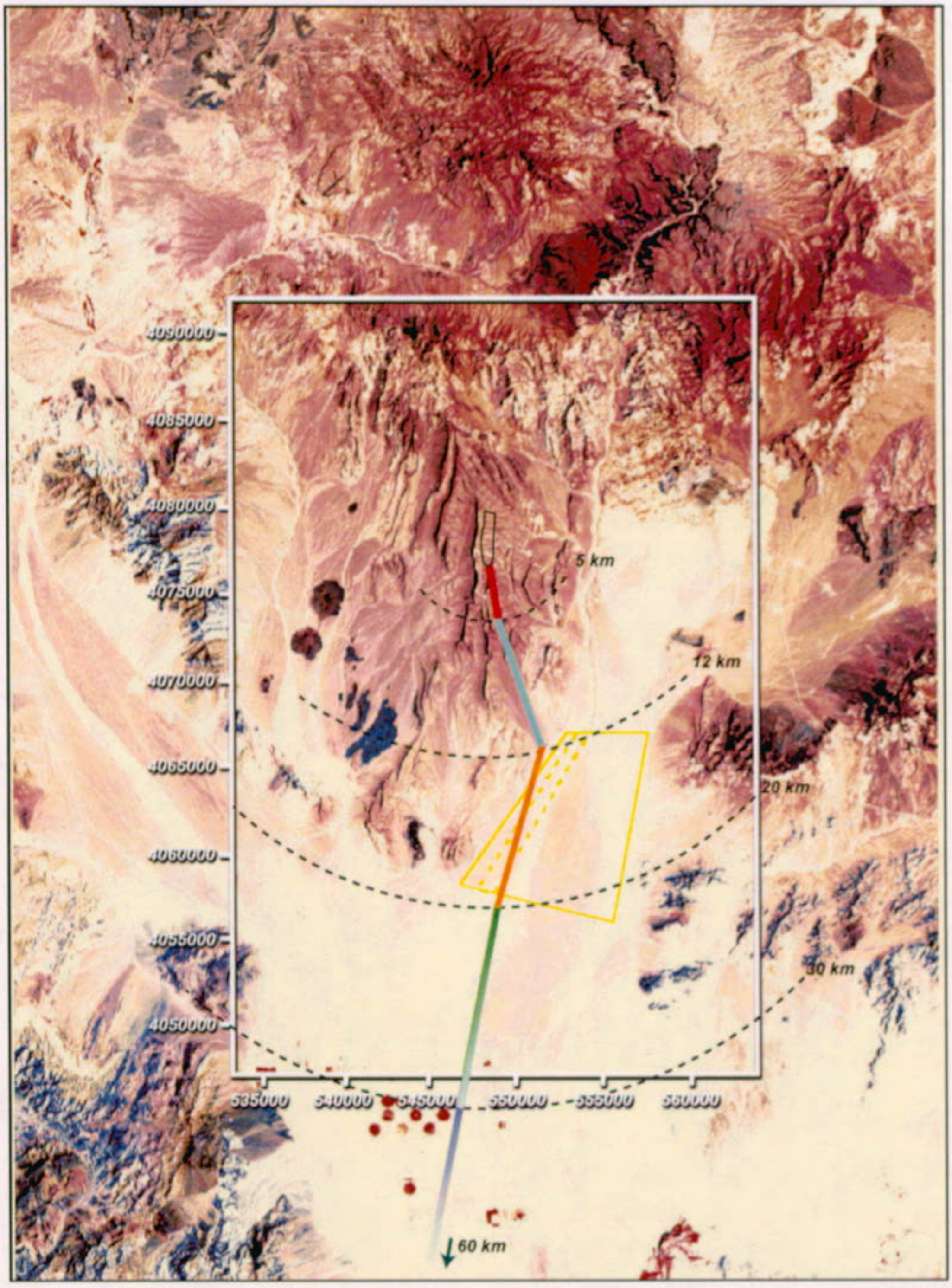
abq0063G136

Figure 3.8-14. Four Source Regions in the Saturated Zone under the Potential Repository Footprint



abq0063G162

Figure 3.8-15. Conceptualization of the One-Dimensional Saturated Zone Transport Model



abq0063G135

NOTE: Dashed lines indicate radionuclide-collection fences; yellow quadrilateral indicates alluvium area of uncertainty.

Figure 3.8-16. The Yucca Mountain Vicinity Showing the Three-Dimensional Saturated Zone Transport Model Domain, the One-Dimensional Saturated Zone Transport Model Flowtube, the Transport Radionuclide-Collection Fences, and the Alluvium Area of Uncertainty

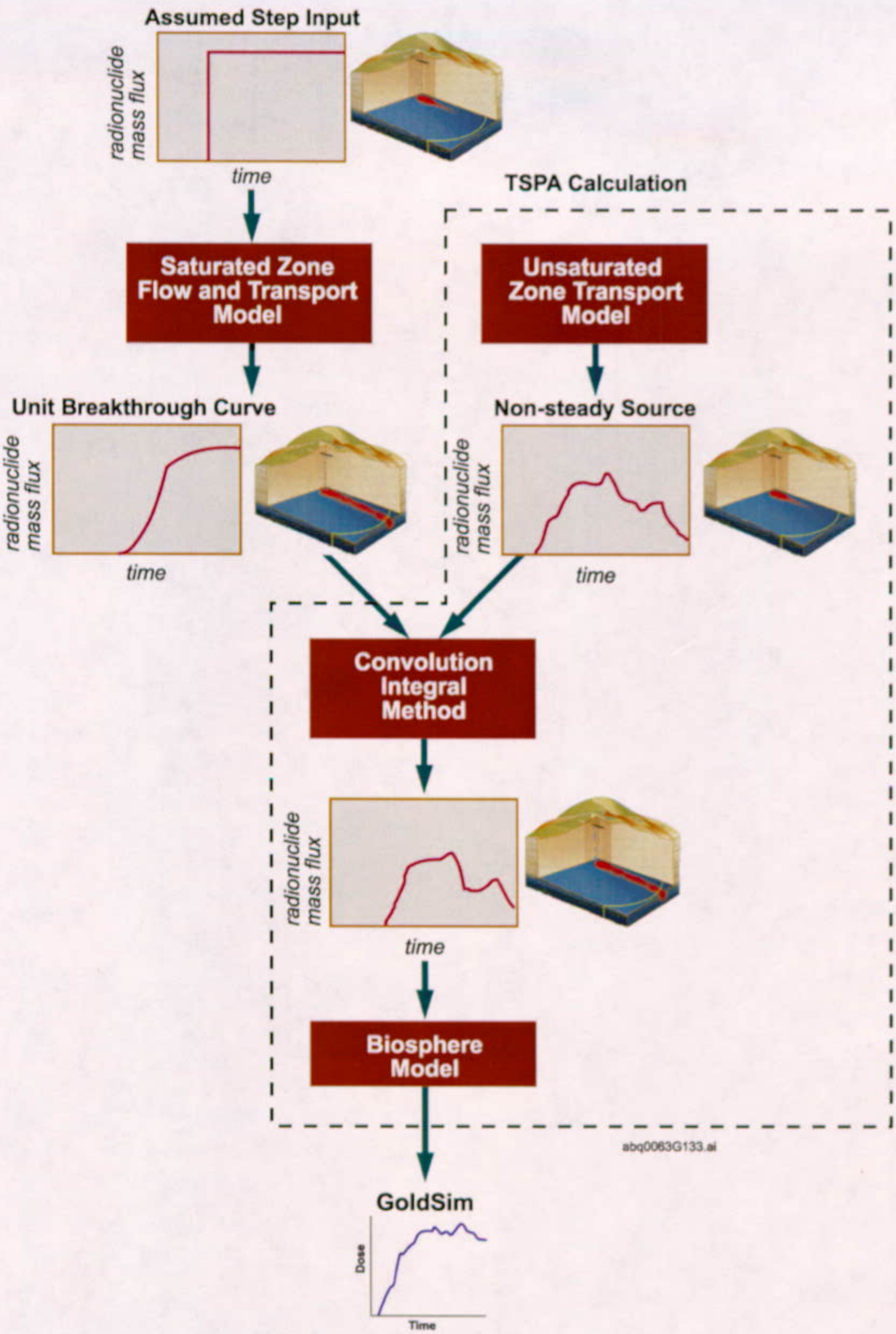


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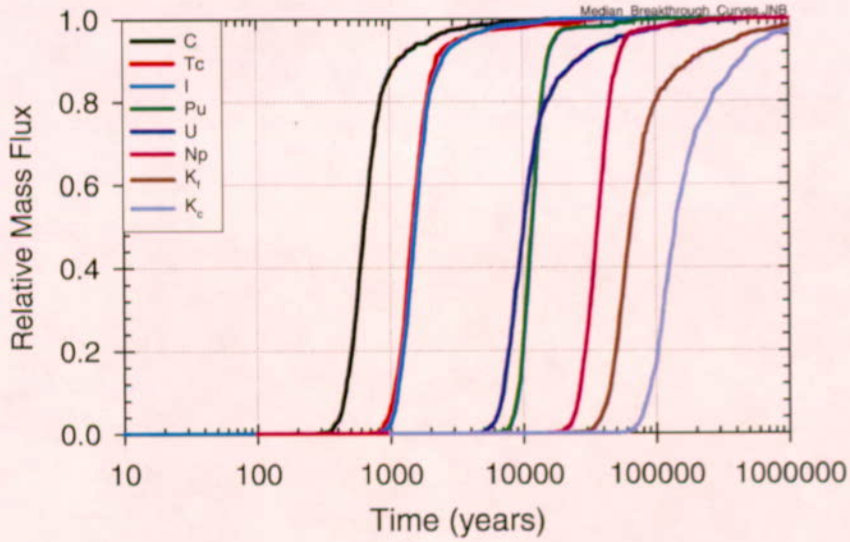
Figure 3.8-17. Flow Chart of the Implementation of the Three-Dimensional Saturated Zone Transport Model into the Total System Performance Assessment-Site Recommendation

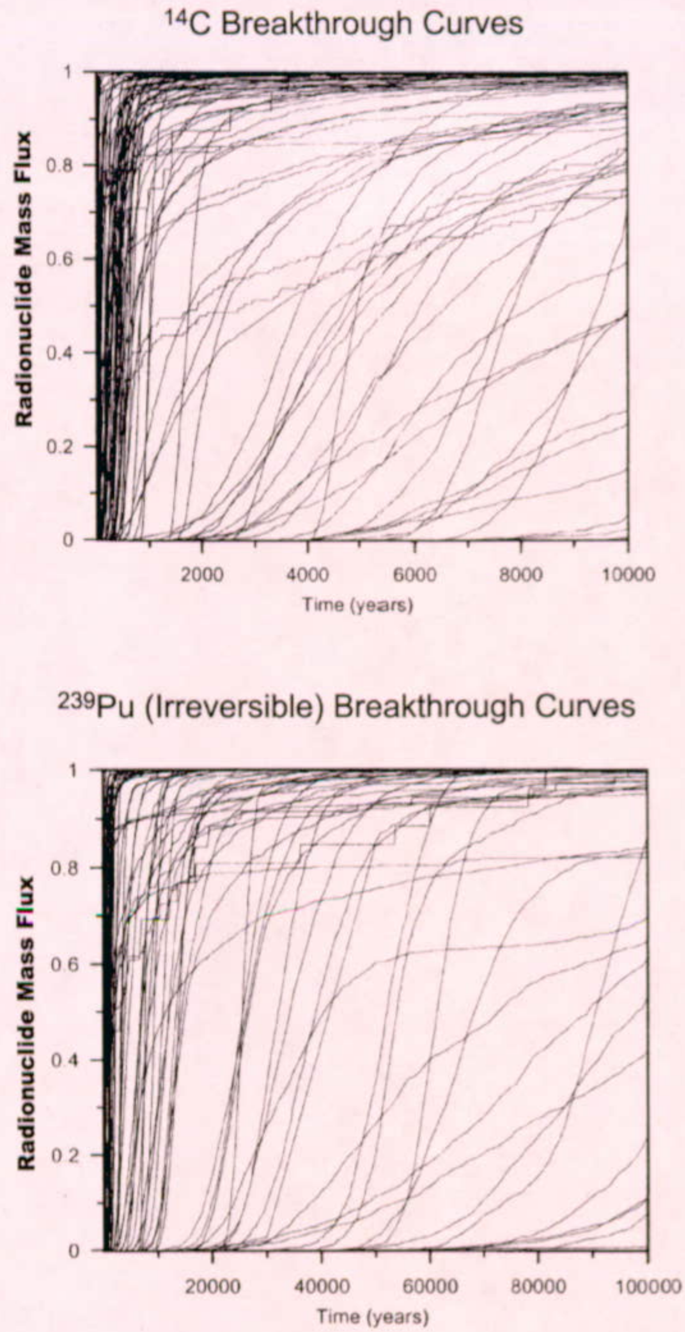


DTN: SN0004T0501600.004 [149288]

abq0063G157

NOTE: Breakthrough curves are calculated using the median values of all parameters

Figure 3.8-18. Breakthrough Curves Calculated by the Three-Dimensional Saturated Zone Transport Model for the Eight Radionuclide Classes Using Median Parameter Values

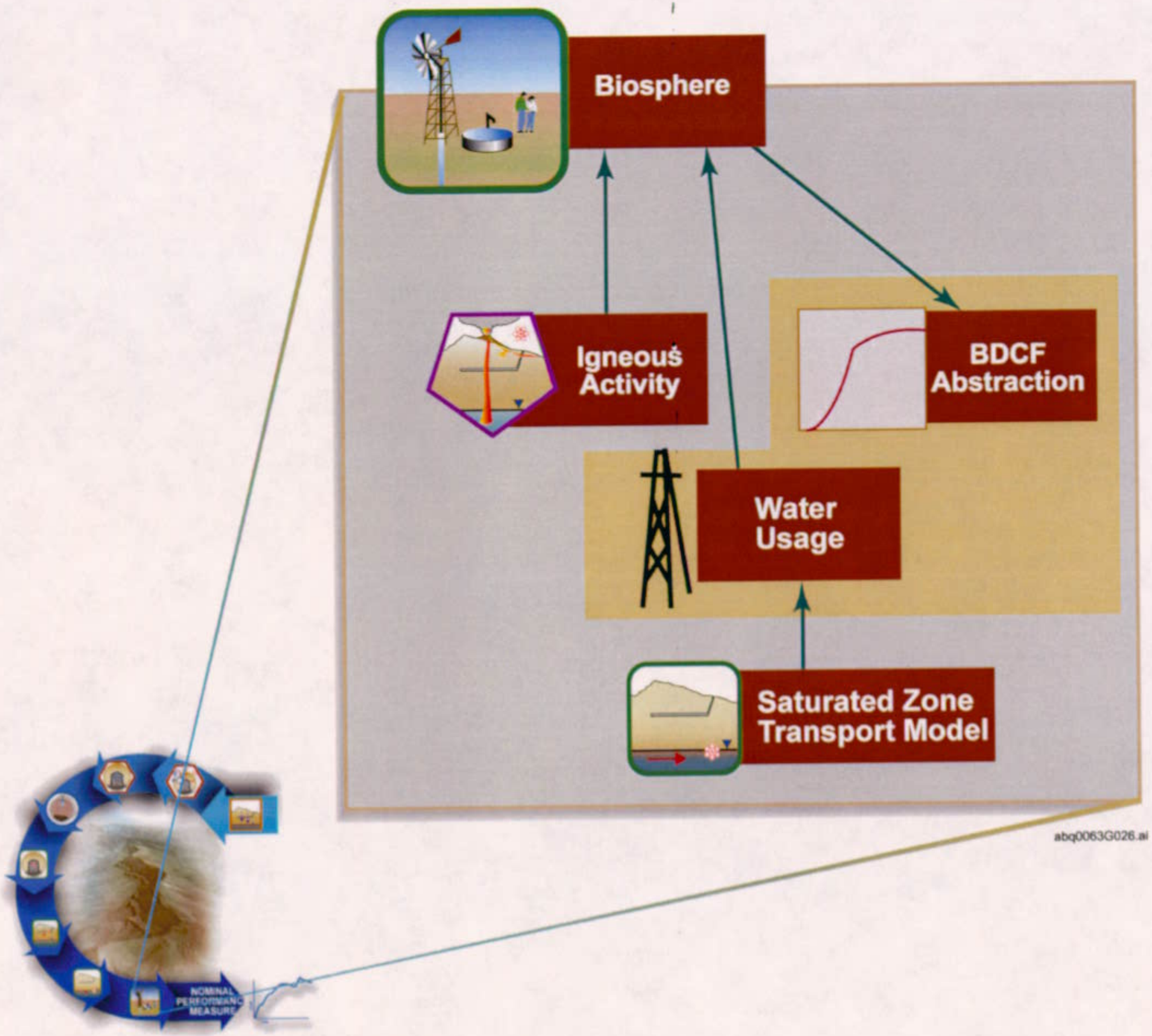


abq0063g161.ai

DTN: SN0004T0501600.004 [149288]

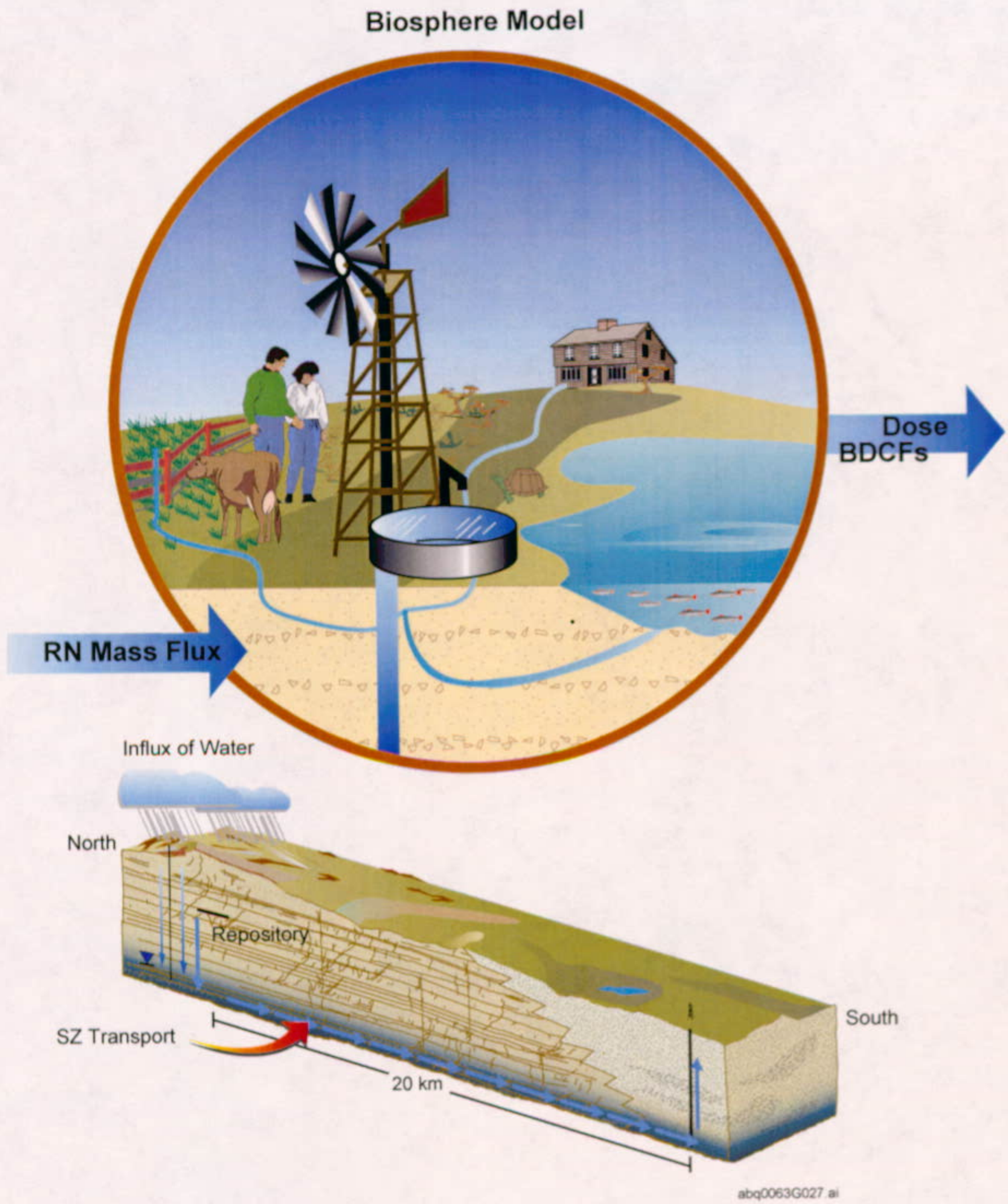
abq0063G161

Figure 3.8-19. Breakthrough Curves Calculated by the Three-Dimensional Saturated Zone Transport Model for 100 Probabilistic Realizations for (a) Carbon and (b) Plutonium Irreversibly Associated with Colloids



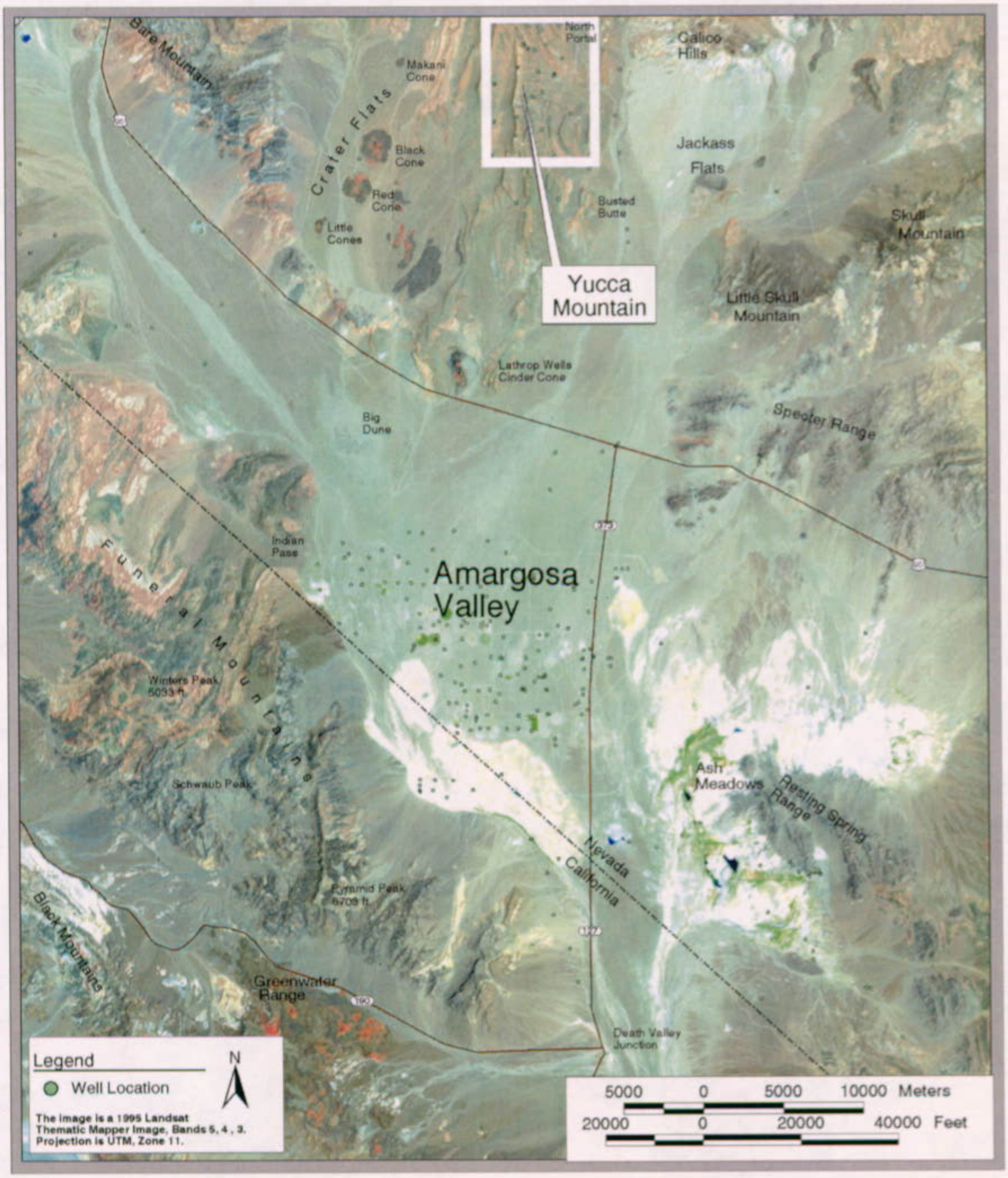
abq0063G026
NOTE: Index figure in lower left is same as Figure 2.1-6

Figure 3.9-1. Relationship of the Biosphere Component and its Relationship with other Total System Performance Assessment Components



abq0063G027

Figure 3.9-2. Overview of the Biosphere Component



abq0063G308

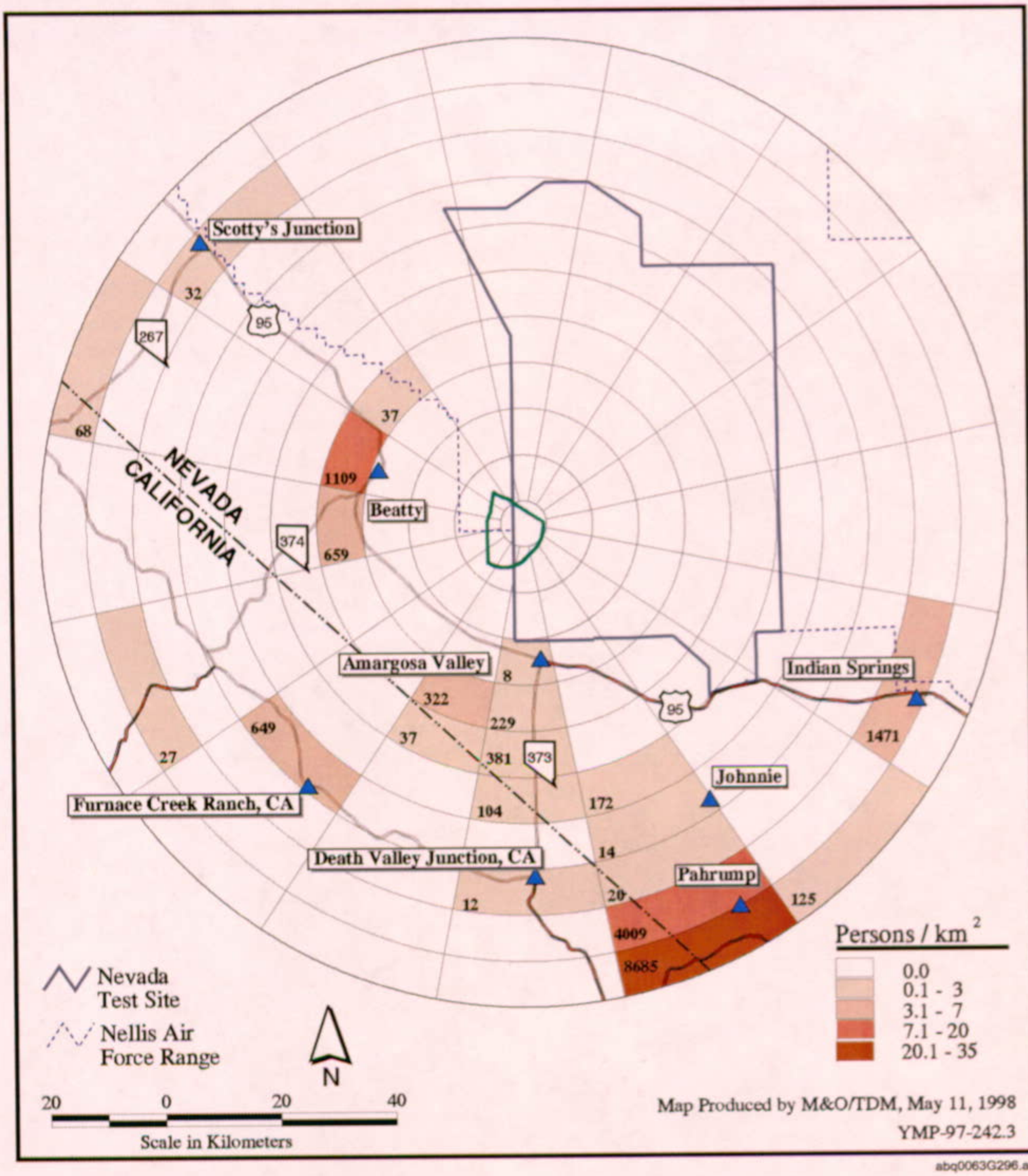
Figure 3.9-3. Map of Yucca Mountain and the Amargosa Valley Region

abq0063G238.ai

abq0063G238

NOTE: Upper Photograph Shows the General Store in the Community. Lower Photograph Shows Central-Pivot Irrigation of an Alfalfa Field

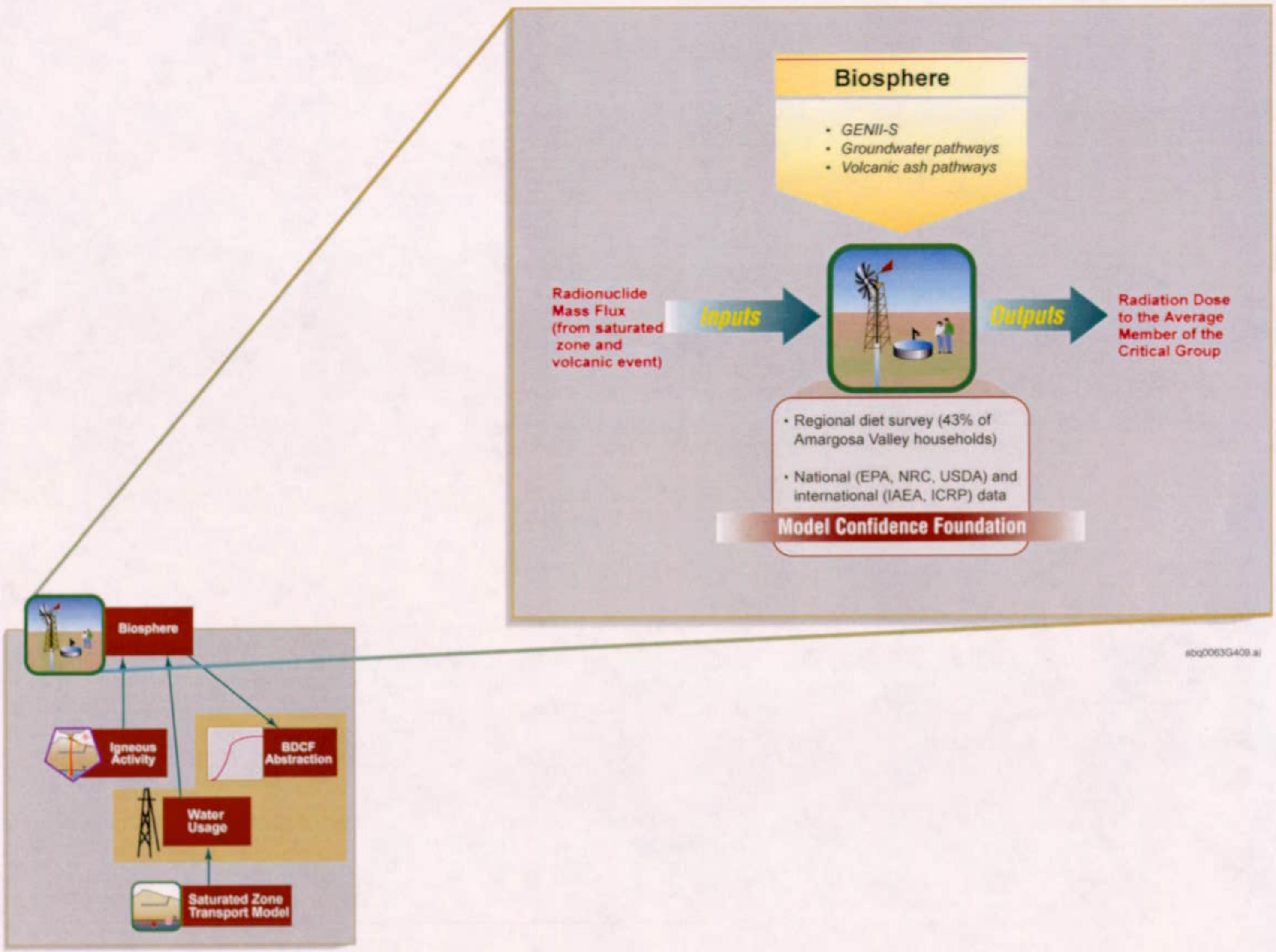
Figure 3.9-4. Present-Day Biosphere in the Amargosa Valley



abq0063G296

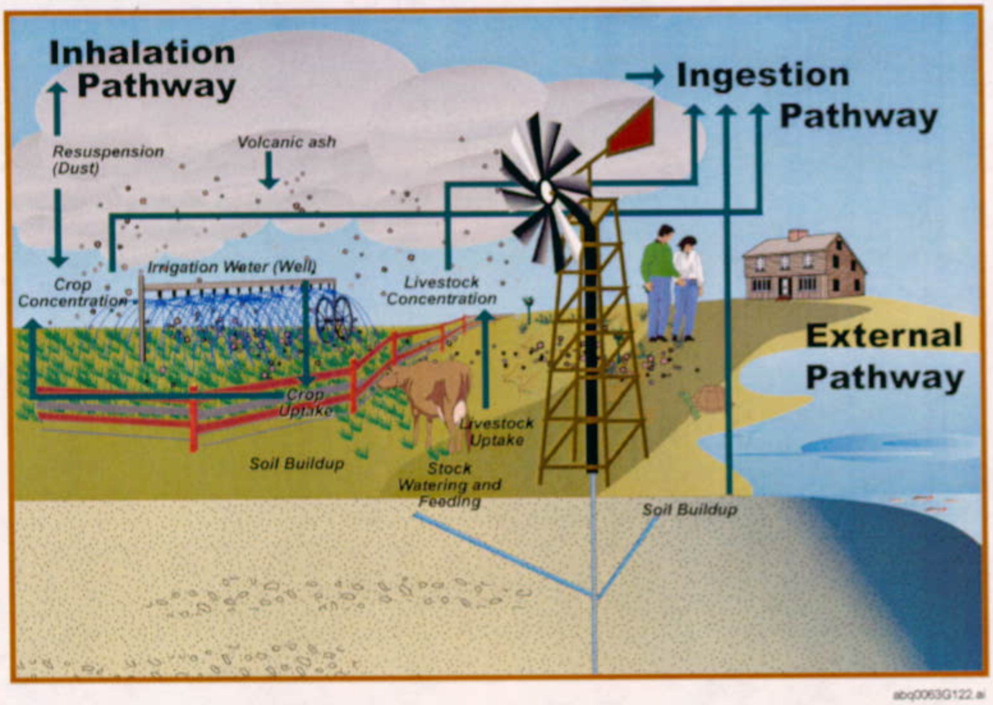
Source: DOE 1998 [100550], Volume 3

Figure 3.9-5. Map Showing the Number of Permanent Inhabitants in the Area of the Regional Food and Water Consumption Survey



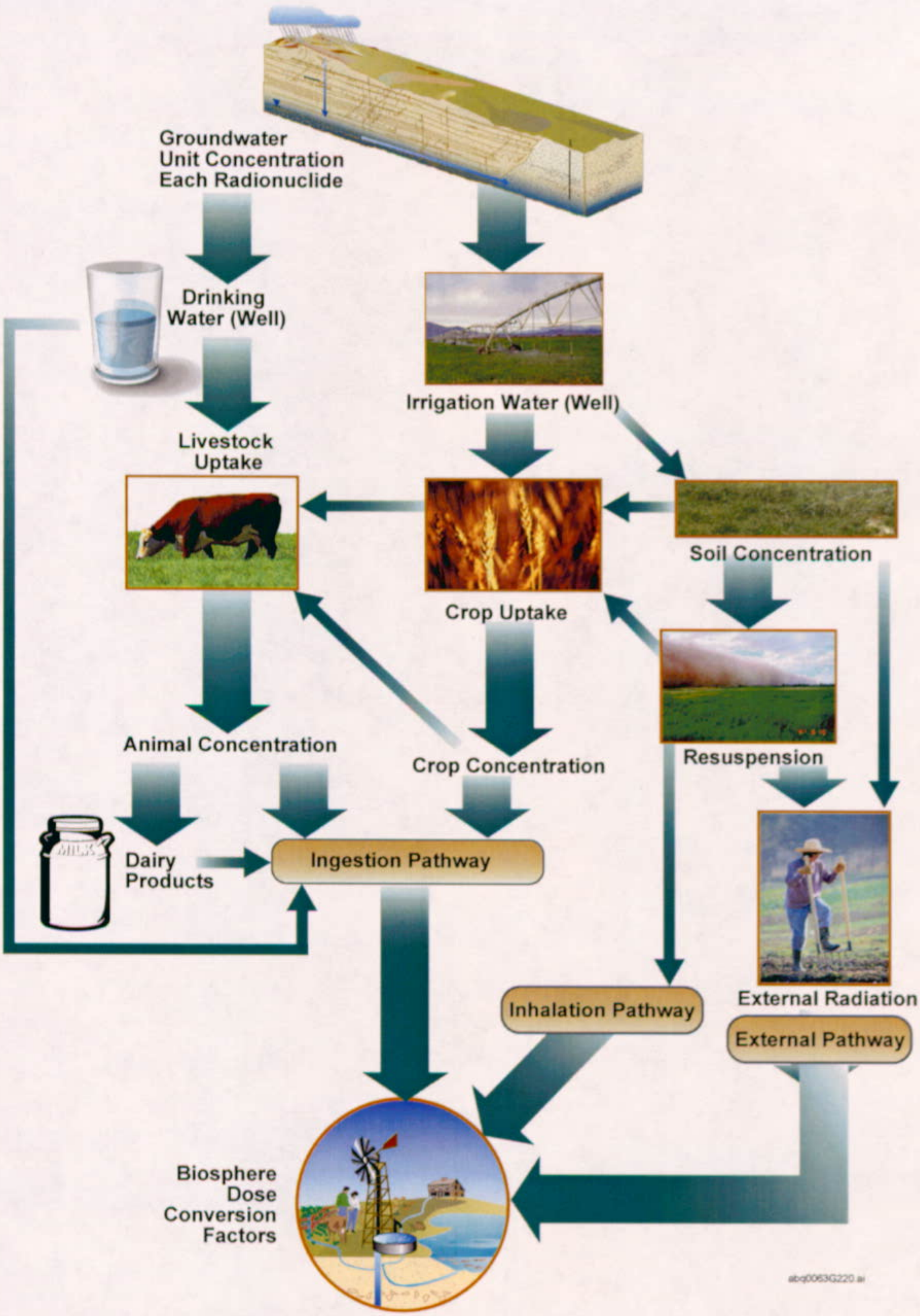
abq0063G409

Figure 3.9-6. Connections between the Biosphere Component and other Total System Performance Assessment Components



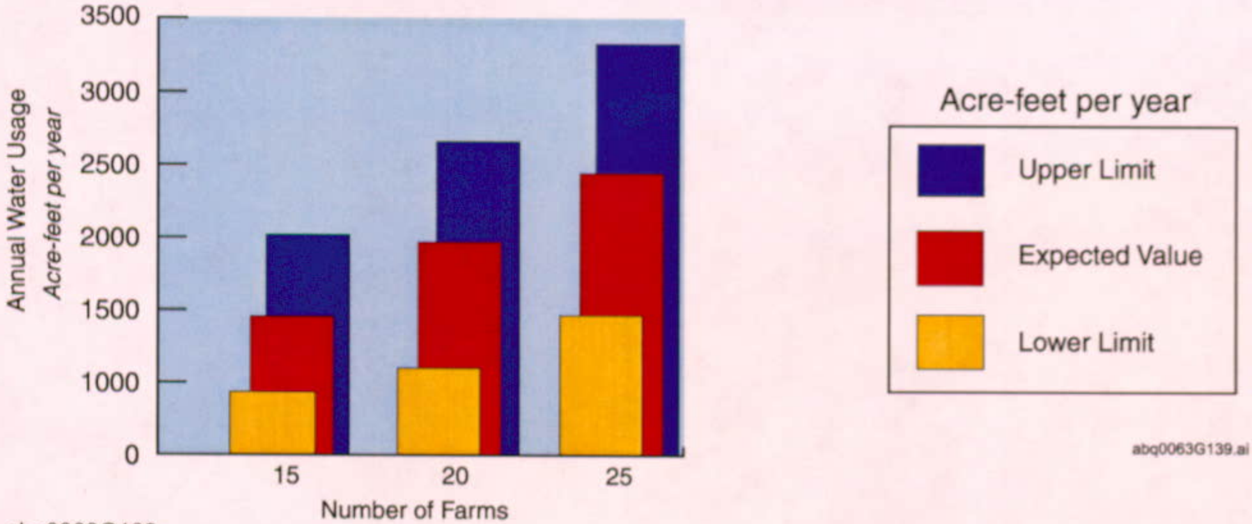
abq0063G122

Figure 3.9-7. Conceptual Illustration of Processes Considered in the Biosphere Model



abq0063G220

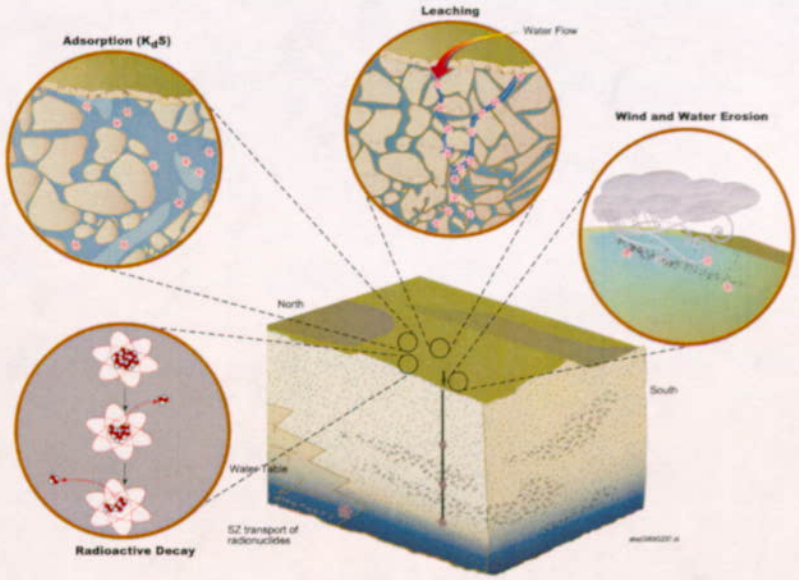
Figure 3.9-8. Diagram of the Pathways Modeled in the Biosphere Model



abq0063G139

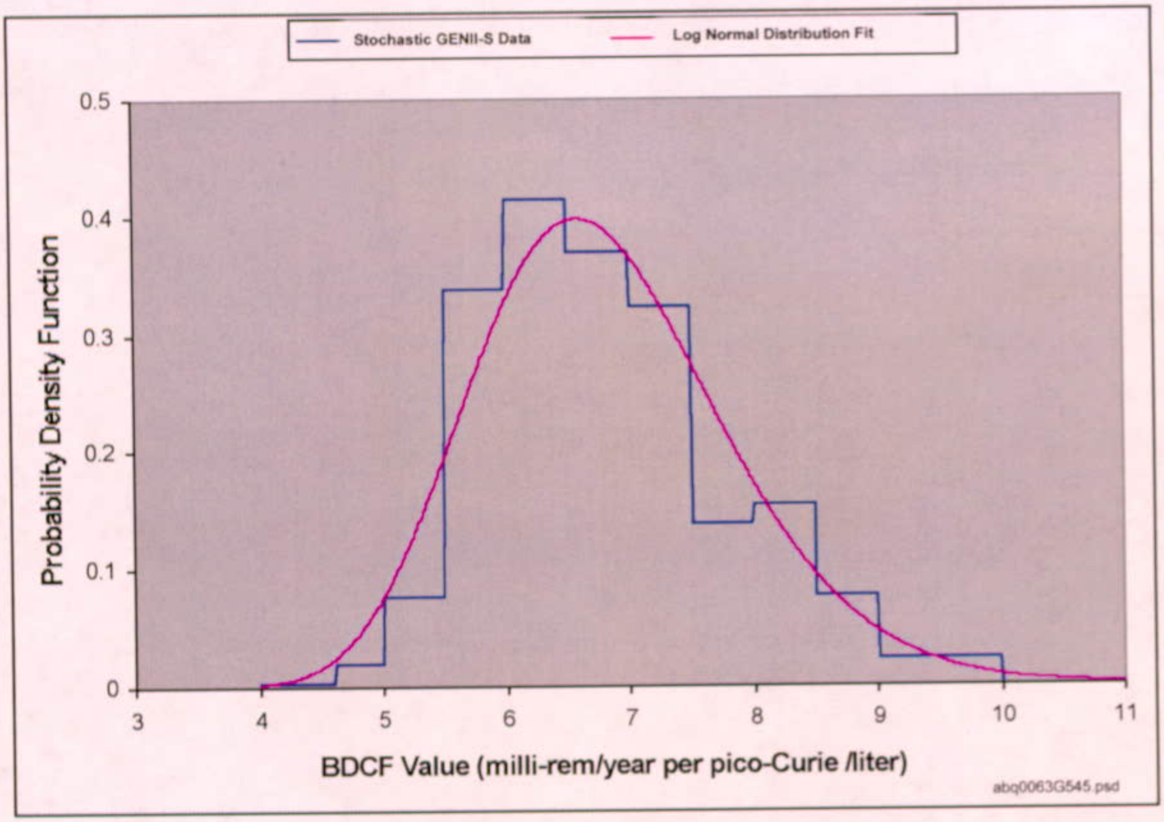
DTN: MO0003SPASGU01.003 [151075]

Figure 3.9-9. Water Usage Volume of a Farming Community in Amargosa Valley as a Function of the Number of Farms



abq0063G237

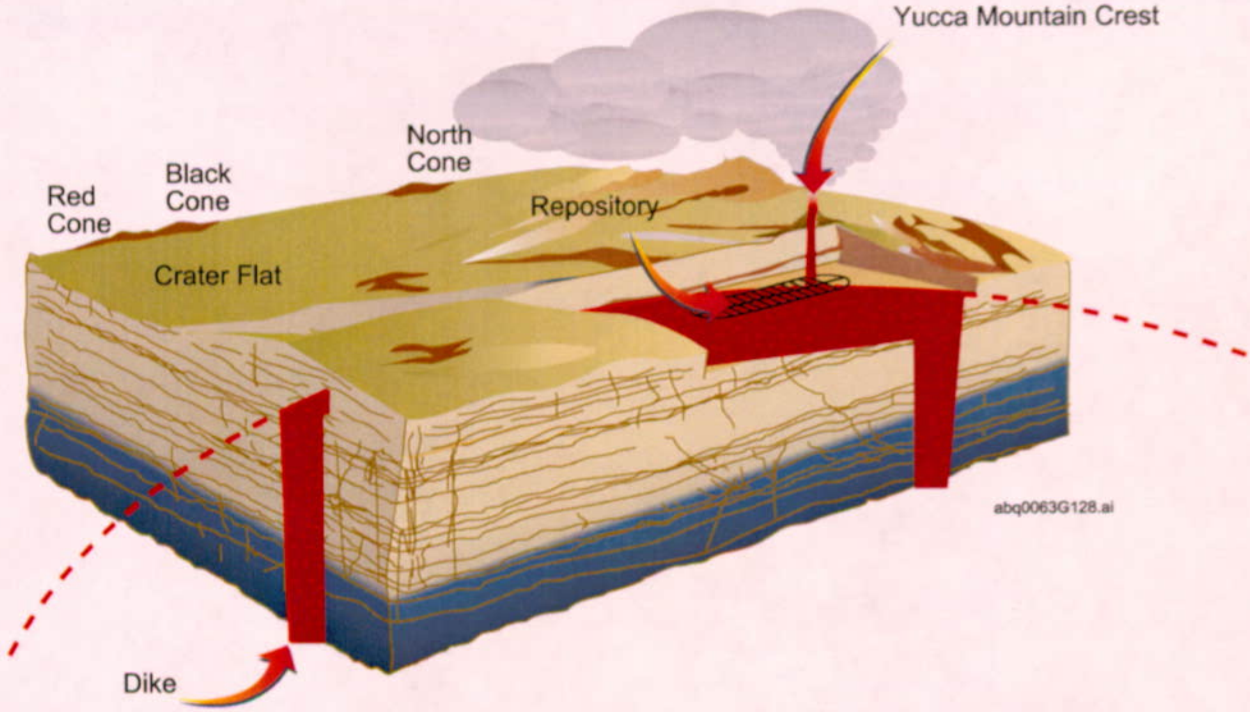
Figure 3.9-10. Conceptualization of Processes Important to Buildup of Radionuclides in Soil



abq0063G545

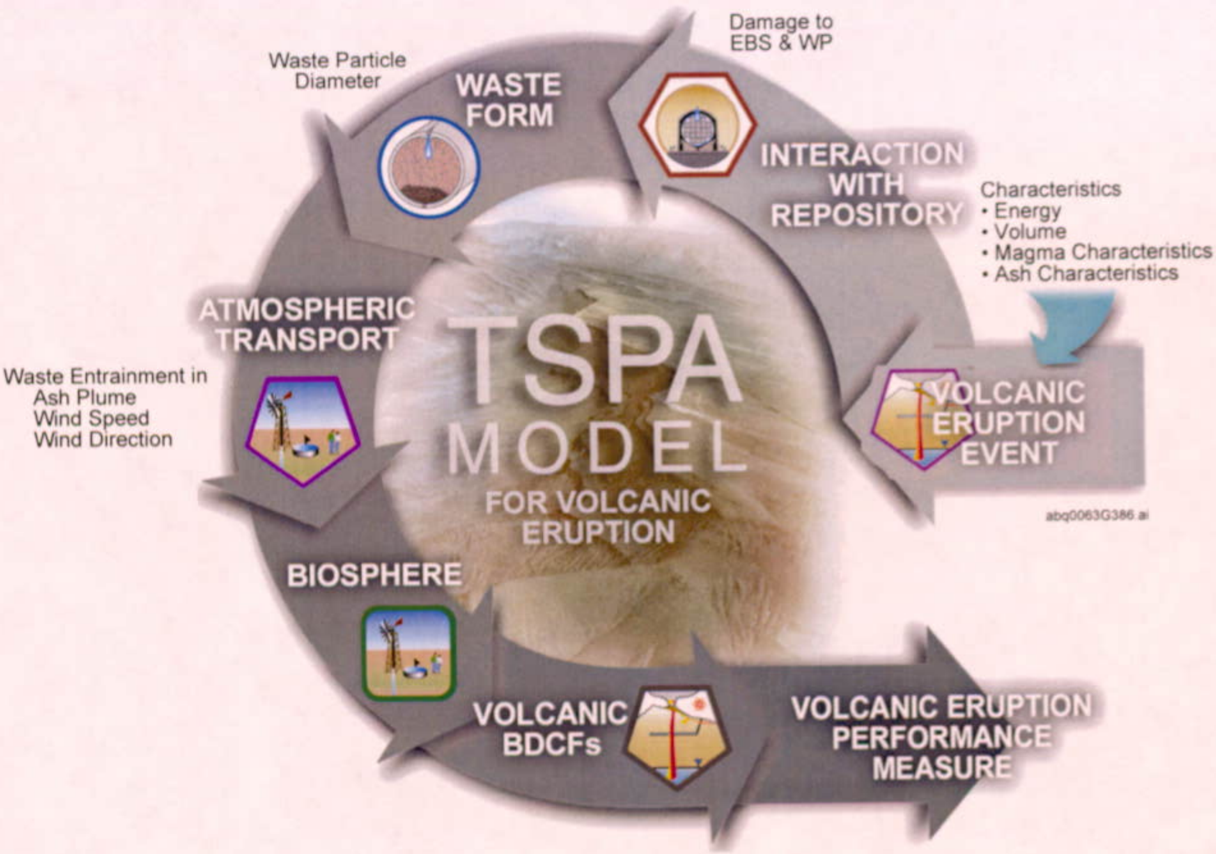
NOTE: With the Optimally Fitted Log-Normal Distribution Data for the Histogram Are in DTN: MO0004SPABDCFS.001 [148923]; the Fitted Curve Is in DTN: MO0003SPAABS08.004 [148453]

Figure 3.9-11. Histogram of the Biosphere Dose Conversion Factor for ^{237}Np



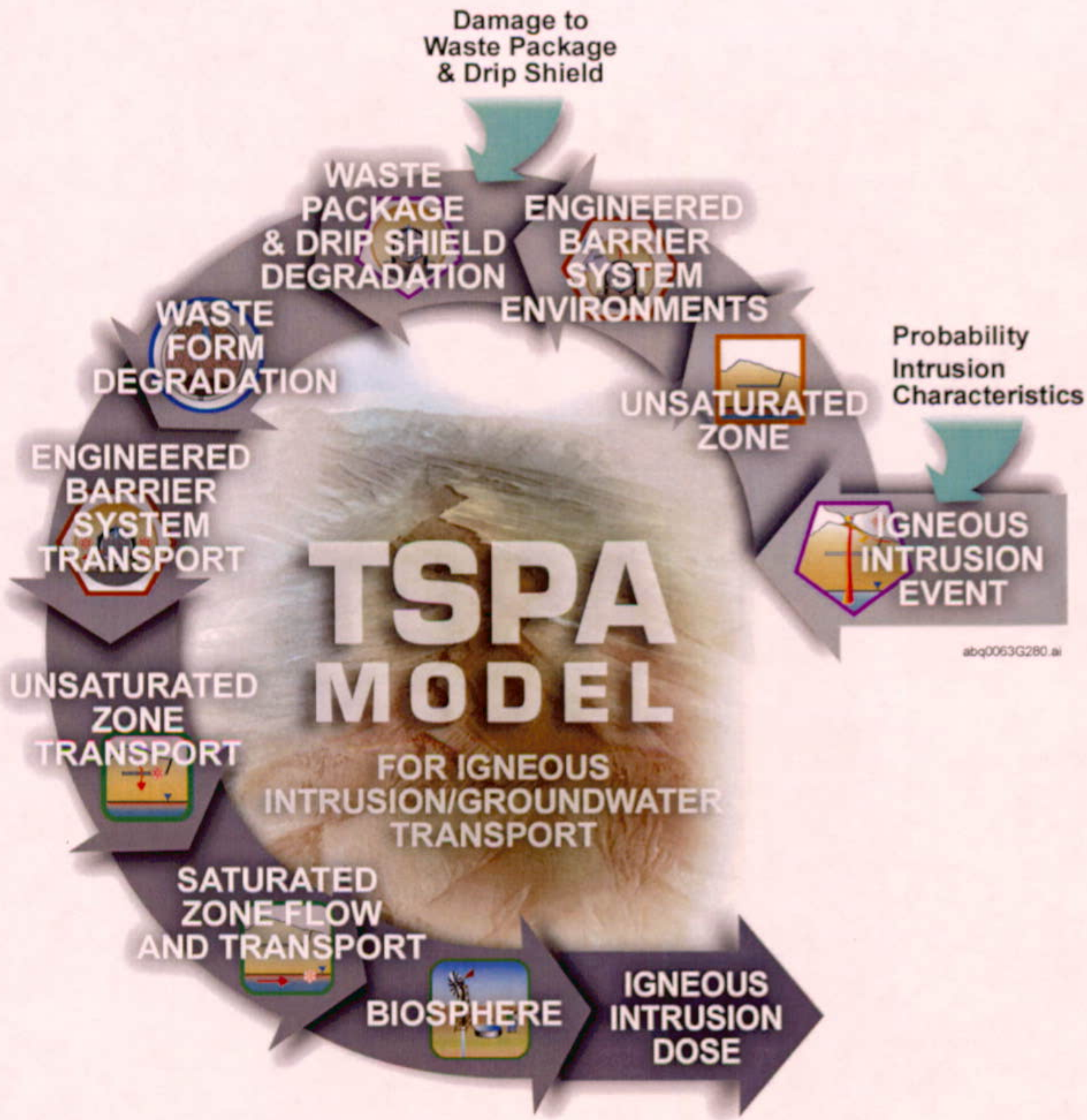
abq0063G128

Figure 3.10-1. Schematic Illustration of Hypothetical Igneous Activity at Yucca Mountain



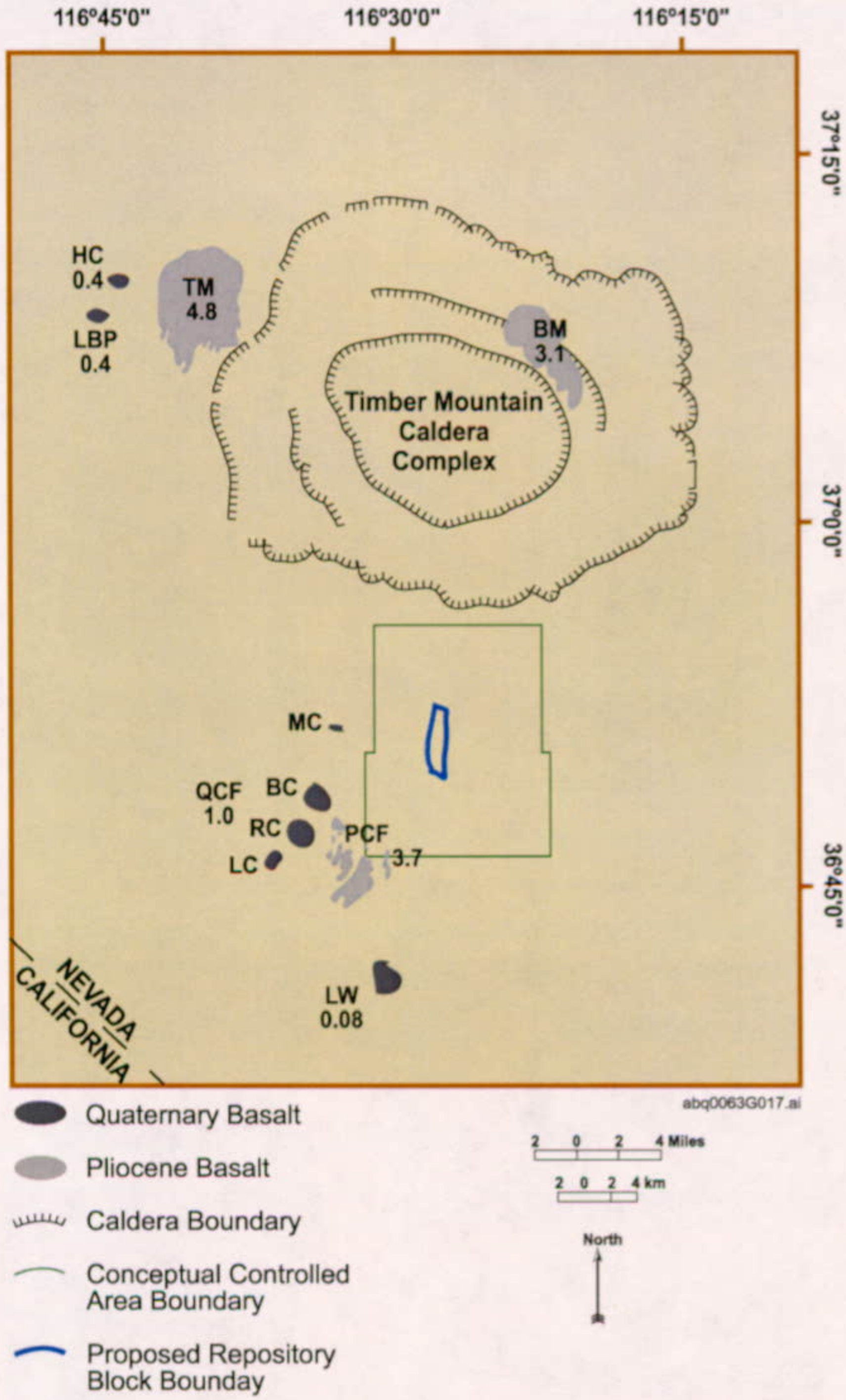
abq0063G386

Figure 3.10-2. Total System Performance Assessment Model Components of the Volcanic Eruption Scenario



abq0063G280

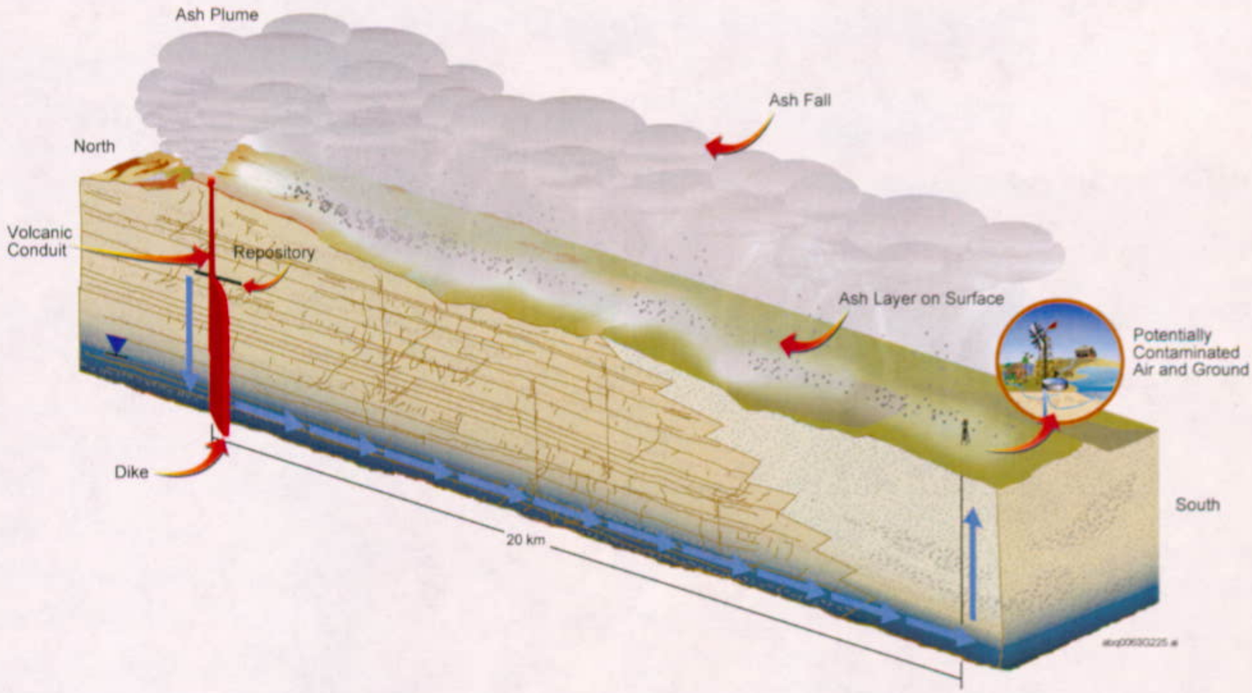
Figure 3.10-3. Total System Performance Assessment Model Components of the Igneous Intrusion Groundwater Transport Scenario



Numbers by each volcano indicate approximate age in millions of years. TM: Thirsty Mesa, PCF: Pliocene Crater Flat, BM: Buckboard Mesa, QCF: Quarternary Crater Flat (MC: Makani Cone, BC: Black Cone, RC: Red Cone, LC: Little Cones), HC: Hidden Cone, LBP: Little Black Peak, LW: Lathrop Wells.

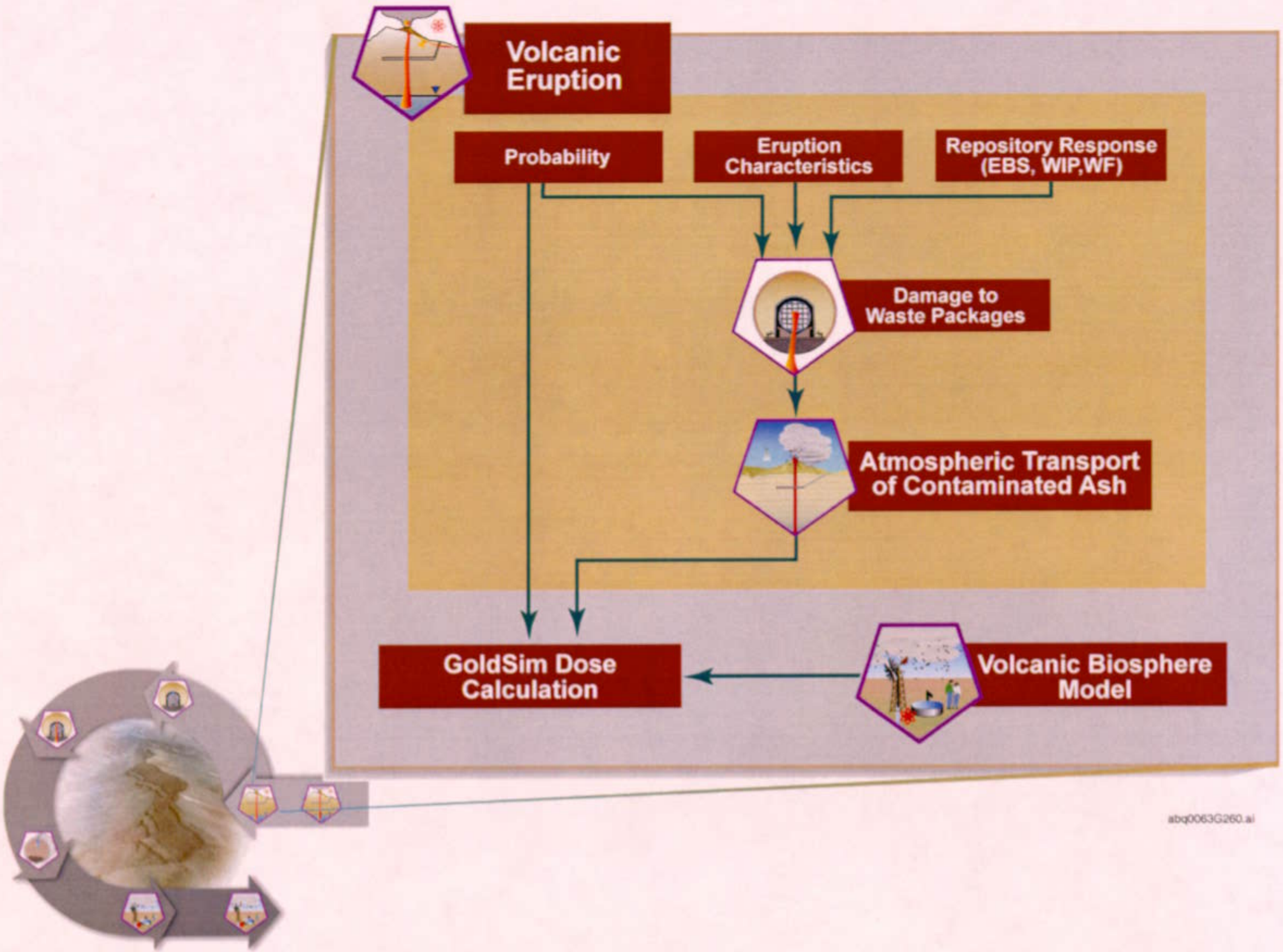
abq0063G017

Figure 3.10-4. Location and Age of Post-Miocene (<5.3 m.y.) Volcanoes (or Clusters where Multiple Volcanoes have Indistinguishable Ages) in the Yucca Mountain Region



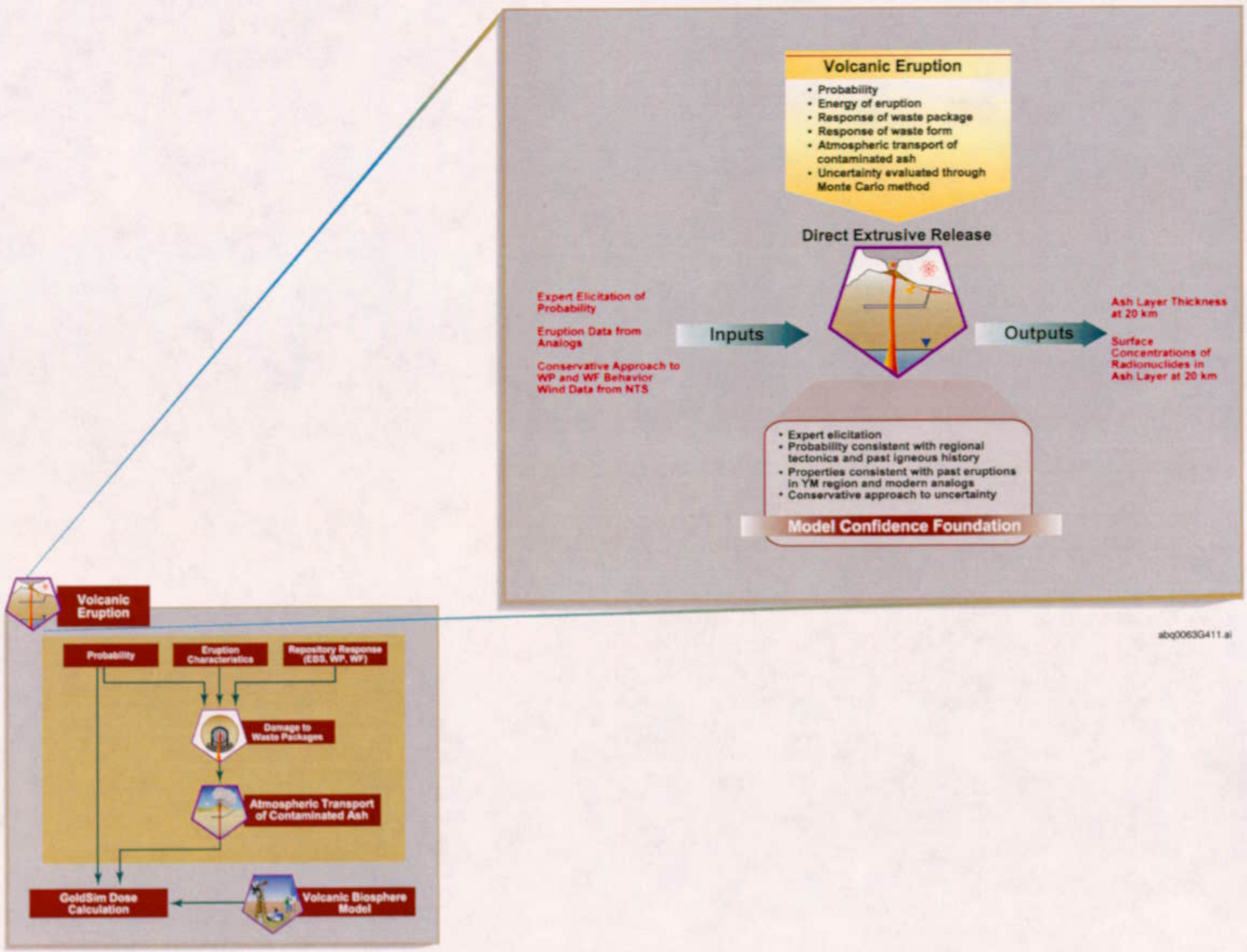
abq0063G225

Figure 3.10-5. Schematic Representation of a Volcanic Eruption at Yucca Mountain, Showing Transport of Radioactive Waste in an Ash Plume



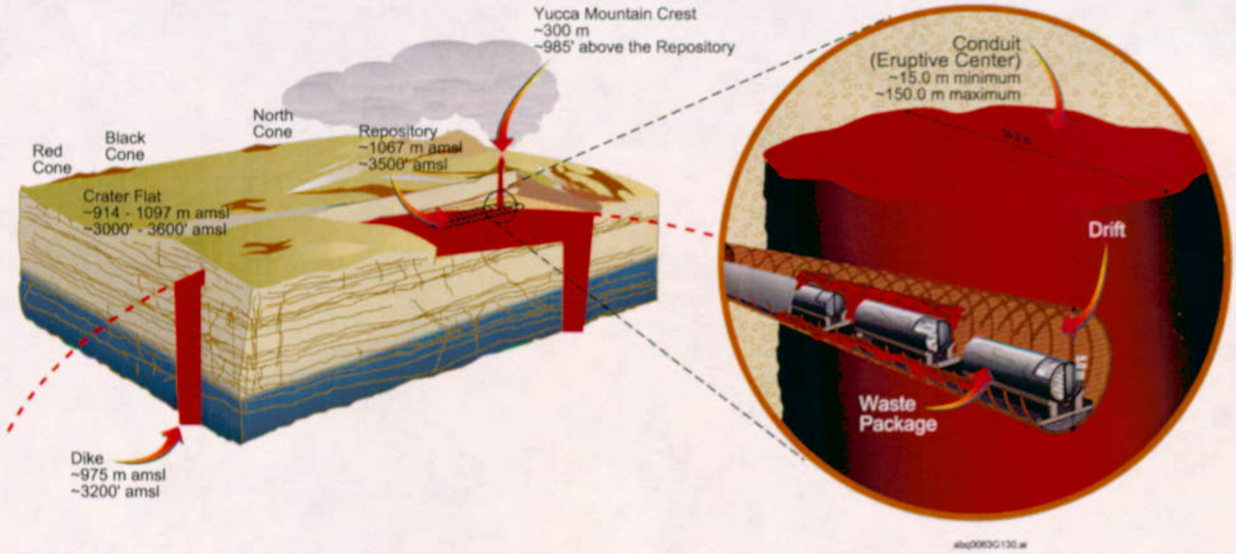
abq0063G260

Figure 3.10-6. Flow of Information within the Total System Performance Assessment Volcanic Eruption Model



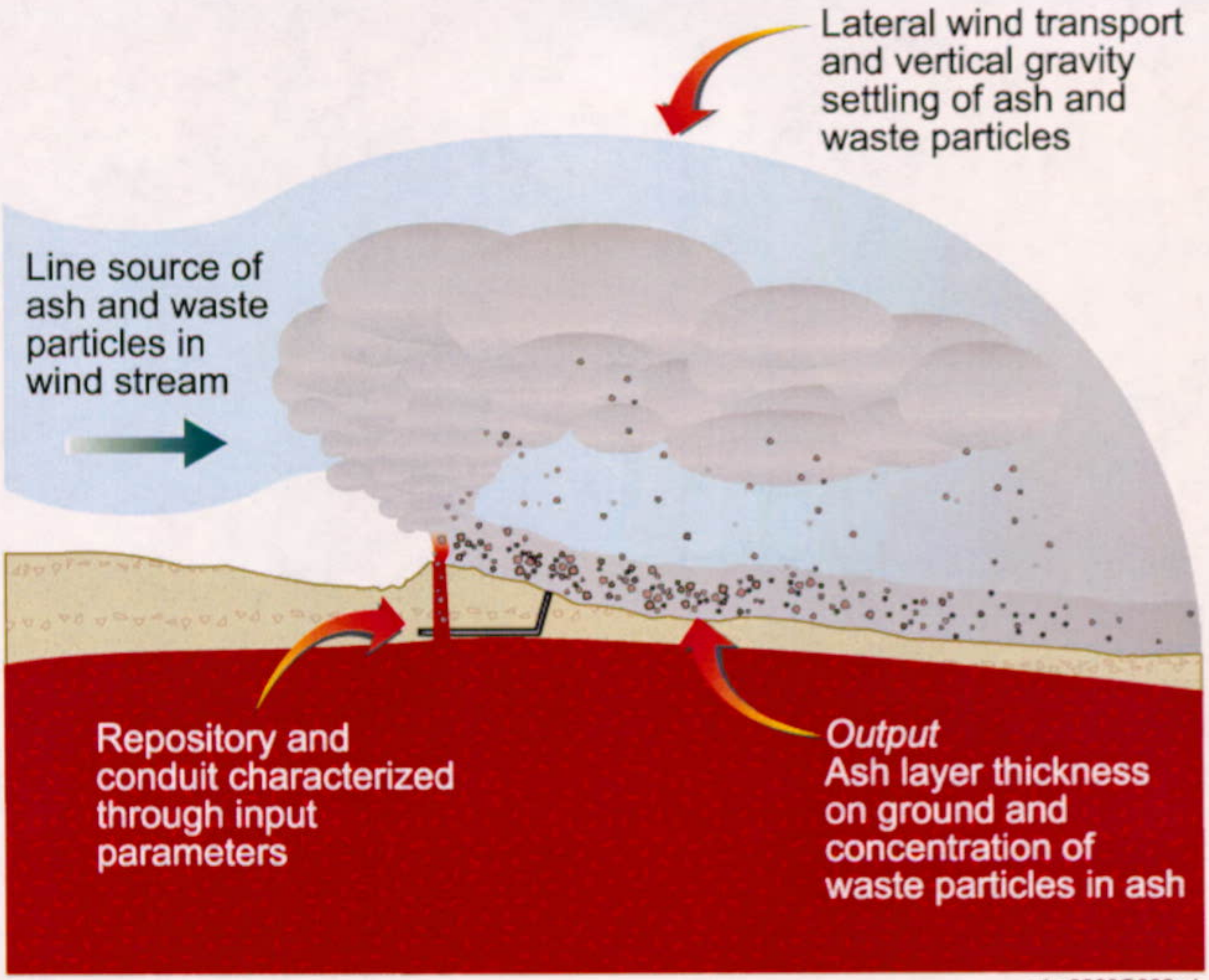
abq0063G411

Figure 3.10-7. Major Inputs and Outputs for the Volcanic Eruption Model



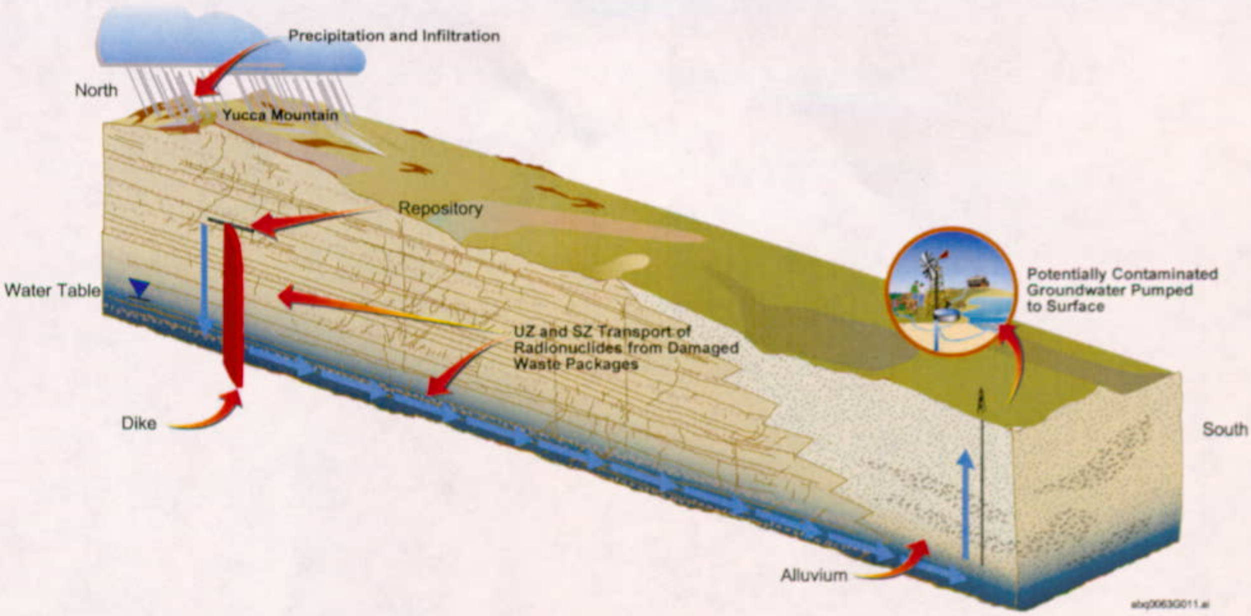
abq0063G0130.ai

Figure 3.10-8. Schematic Diagram of the Intersection of an Igneous Dike and Eruptive Conduit with the Potential Repository



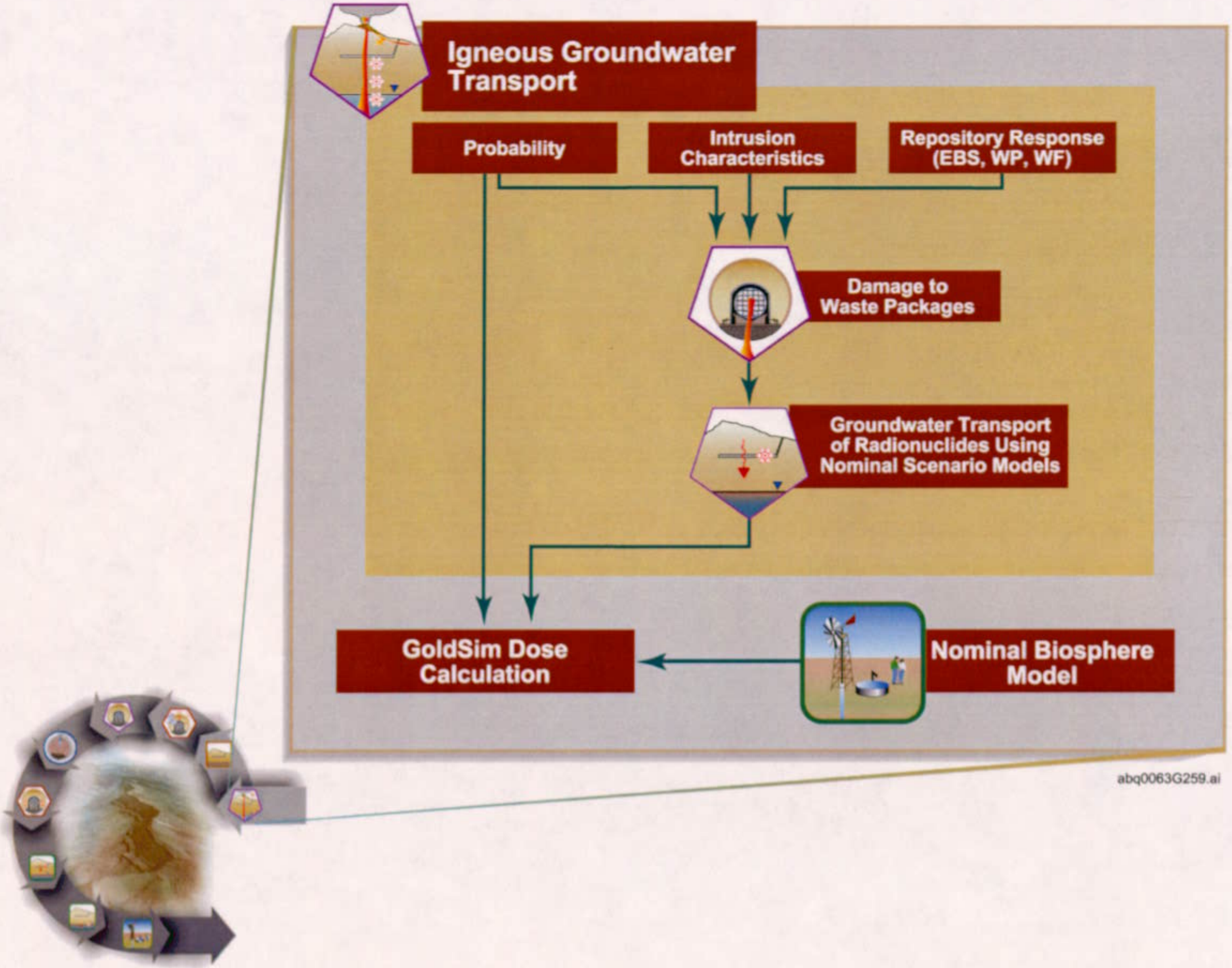
abq0063G002

Figure 3.10-9. Transport of Ash and Radioactive Waste in a Volcanic Ash Plume



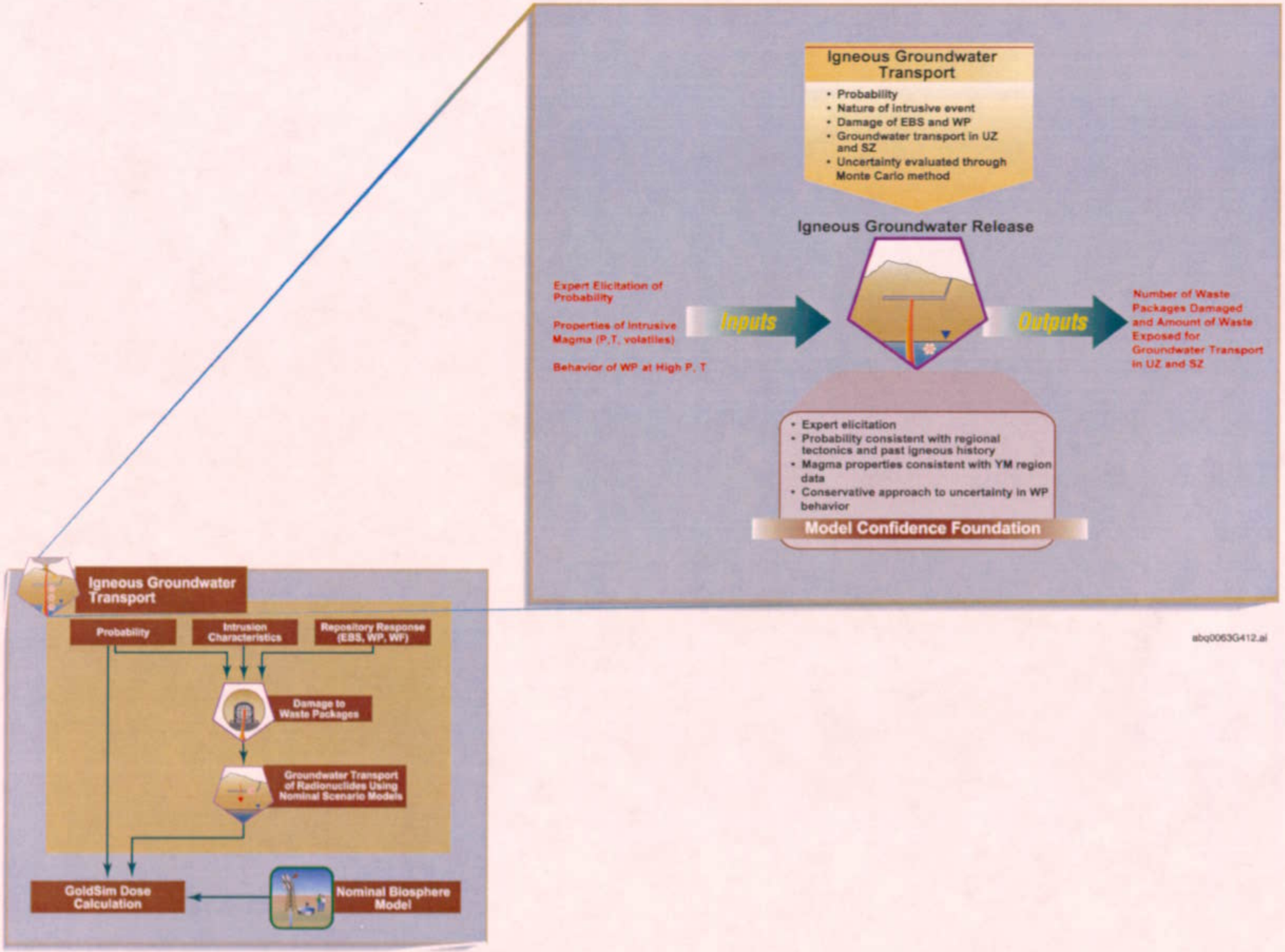
abq0063G011

Figure 3.10-10. Schematic Diagram Showing an Igneous Intrusion at Yucca Mountain and Subsequent Transport of Radionuclides in Groundwater



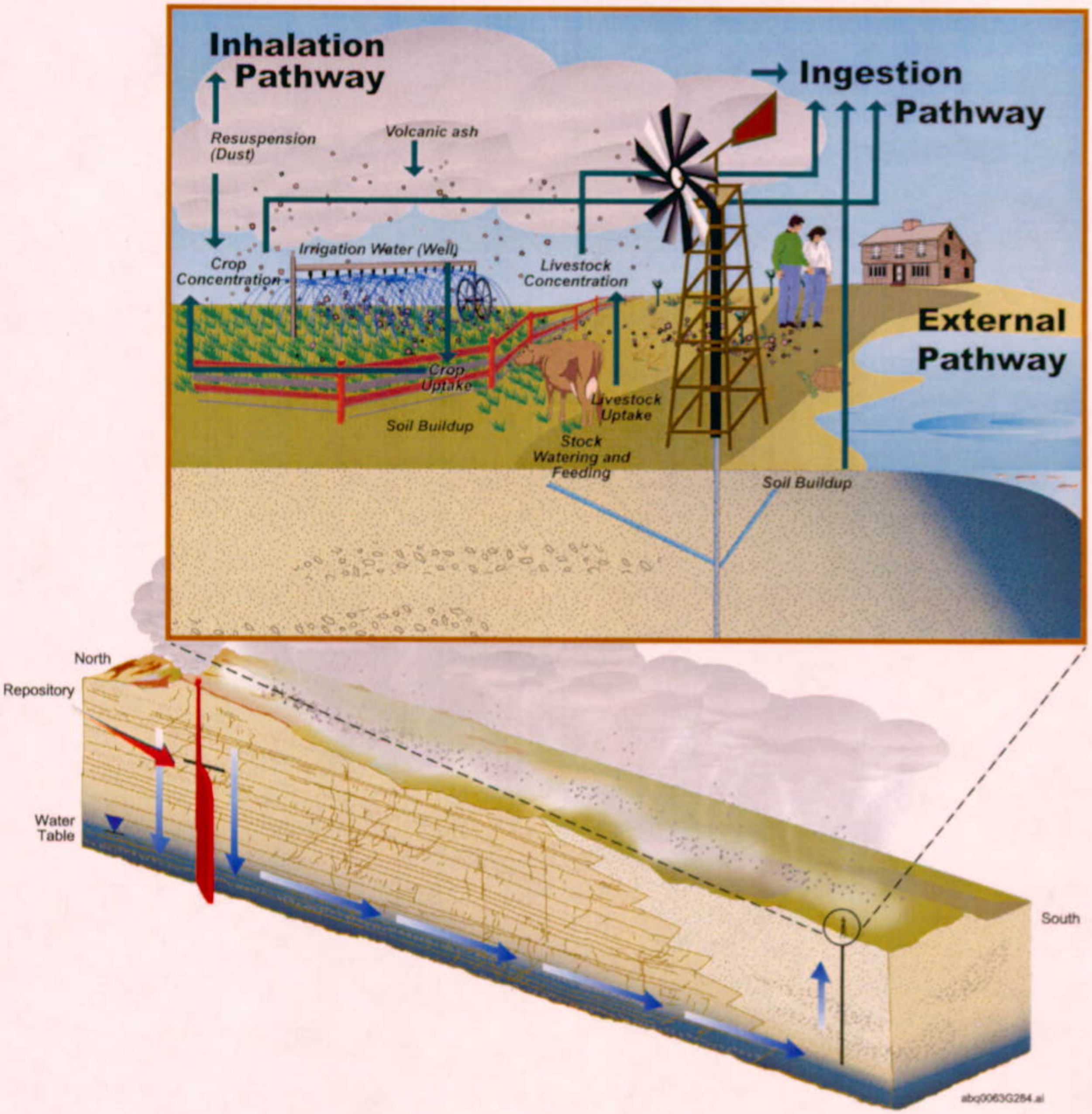
abq0063G259

Figure 3.10-11. Information Flow within the Igneous Intrusion Groundwater Transport Model



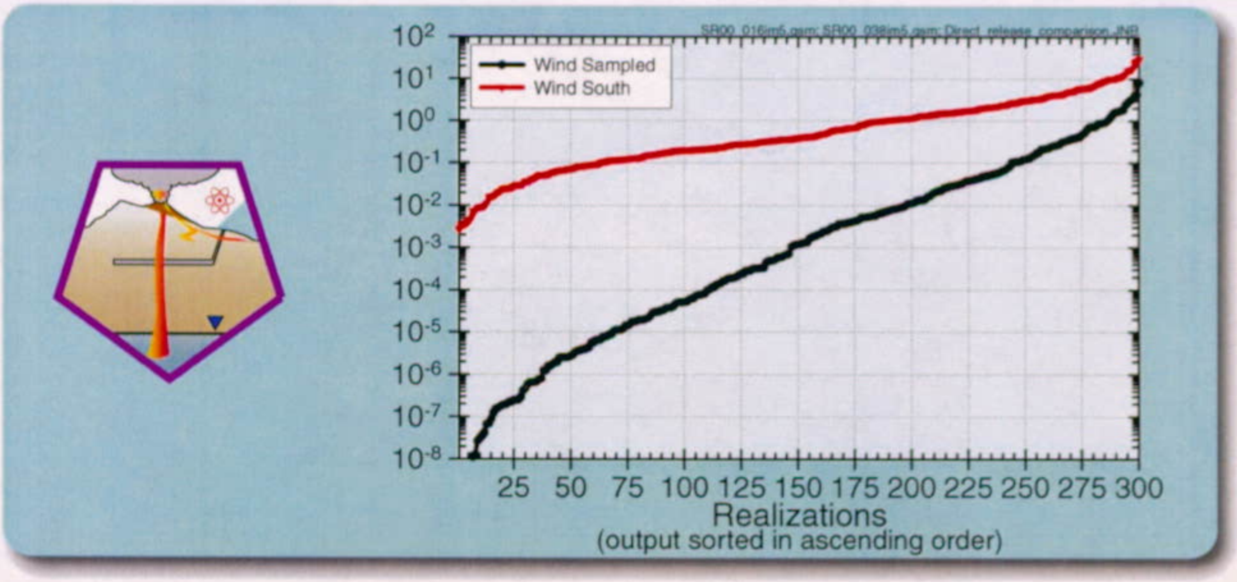
abq0063G412

Figure 3.10-12. Major Inputs and Outputs for the Igneous Intrusion Groundwater Transport Model



abq0063G284.ai

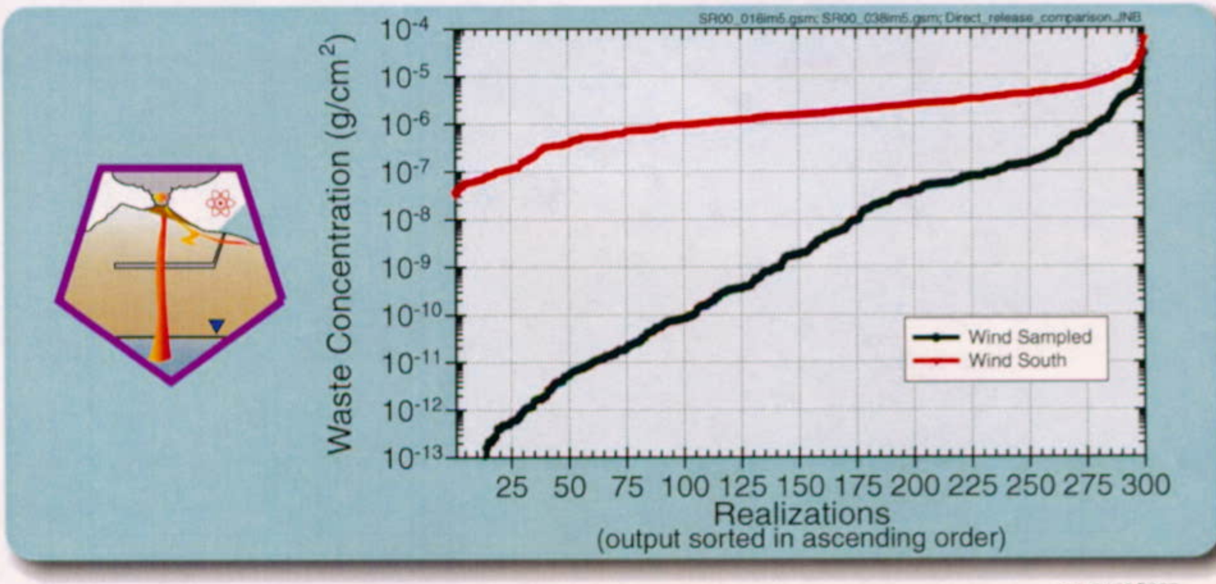
Figure 3.10-13. Schematic Illustration of Possible Pathways for Human Exposure Following Deposition of Contaminated Volcanic Ash



abq0063G606

abq0063G606

Figure 3.10-14. Calculated Ash Layer Thicknesses at 20 km from the Eruption



abq0063G607

Figure 3.10-15. Calculated Waste Concentrations on the Ground Surface 20 Kilometers from the Potential Repository

4. PERFORMANCE ANALYSES

The total system performance analyses of the postclosure period for the potential Yucca Mountain repository are presented in this section. The previous sections have detailed both the approach to the analyses and the subsystem components used in the analyses. Appendix E summarizes the primary source information to the TSPA-SR. This TSPA was conducted according to the guidance from proposed 10 CFR Part 63 (64 FR 8640 [101680]), and 40 CFR 197 and the guidance and criteria specified in proposed 10 CFR Part 963 (64 FR 67054 [124754]) of the Site Suitability Guidelines.

The nominal performance evaluation is presented in Section 4.1 and includes a systematic analysis of the contribution of each component to the total system dose. The nominal performance case analyses evaluate the total repository system in the absence of low-probability disruptive events.

Section 4.2 provides a discussion of the disruptive performance analyses. The igneous disruption scenario described in this section covers both the effect of direct release of radionuclides due to an igneous intrusion through the repository leading to an eruption and the effect of indirect release of radionuclides due to igneous intrusion damaging a number of waste packages at the repository level. Although the igneous disruptive scenario is a very low probability event, it is analyzed within the TSPA to show its potential consequence.

As is required by the proposed 40 CFR Part 197 (64 FR 46976 [105065]) and the proposed 10 CFR Part 963 (64 FR 67054 [124754]), Section 4.3 provides an evaluation of the combined nominal and disruptive event scenarios to show overall performance.

Section 4.4 presents the stylized scenario as required by the proposed 10 CFR Part 963 (64 FR 67054 [124754]) to evaluate an assumed human intrusion occurrence. The general parameters for the analysis are defined by the regulation, but additional models and assumptions are required to evaluate the scenario. The alternative scenarios considered by the U.S. Environmental Protection Agency in proposed 40 CFR Part 197 (64 FR 46976 [105065]) were not analyzed.

Section 4.5 provides coverage of some of the major potentially disruptive events that have not been included in either the nominal or disruptive events because they were screened out as being either inconsequential or of extremely low probability. Among the events discussed qualitatively in this section are the criticality scenario and the scenario of a rapid increase of groundwater to inundate the repository.

Section 4.6 closes this section with a discussion and analysis of a couple design alternatives. The two cases considered were 1) the reference design with backfill, and 2) the low thermal load design. The discussion in this section focuses on the analyses conducted using the alternative designs and comparing them with the no-backfill reference design.

Appendix G lists the simulations that have been conducted for the Total System Performance Assessment-Site Recommendation (TSPA-SR). For each model case, the appendix provides the run number and run name by which the case is referred to, the basic scenario used, and a description of the run. The appendix also lists the figures illustrating the model case results, which can be found in this report, and finally the Data Tracking Number (DTN) through which the model data can be accessed in the Technical Data Management System's Model Warehouse. The documentation of the base case model is found in *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]).

4.1 TOTAL SYSTEM PERFORMANCE FOR THE NOMINAL SCENARIO

The nominal scenario consists of all relevant FEPs that are expected to occur over the thousands to tens of thousands of years following the construction of the potential repository and emplacement of the waste packages. The nominal scenario is distinguished from the potentially disruptive scenario described in Section 4.2, which includes events and processes (specifically, those related to igneous activity) that have low probability of occurrence over the periods of interest.

Total system performance assessments (TSPAs) are by nature uncertain projections of the possible behavior of the potential repository system. These projections reflect the relevant processes affecting the containment and isolation of radioactive wastes from the biosphere. Uncertainty is explicitly included in the models and the resulting analyses in the form of probability distributions that encompass a range of possibilities. In the results presented, uncertainty in the possible performance is evaluated through the use of these probabilistic analyses.

The nominal scenario consists of models and parameters of the processes described in Section 3, in particular:

- Groundwater flow in the UZ, including infiltration processes at the ground surface, water seepage into drifts, and the effects of climate changes
- The effects of the temporal and spatial evolution of the physical and chemical environments on the engineered barriers (notably the drip shields and waste packages)
- The degradation of the engineered barriers within this range of possible physical and chemical environments
- The physical and chemical environments within the waste packages once the primary containment has been degraded to the point that through-going cracks penetrate the waste package
- The alteration of the waste form within the waste package, whether it is CSNF, DSNF, HLW, or another waste form, such as immobilized plutonium or naval SNF
- The release of dissolved or colloidal radionuclides through the degraded engineered barriers to the host rock
- The transport of dissolved or colloidal radionuclides through the UZ to the water table

- The transport of dissolved or colloidal radionuclides through the SZ to the point where the water is extracted for irrigation or household use, assumed to be 20 km (12 miles) downgradient from the potential repository

- The transport of radionuclides in the biosphere through a range of possible biological pathways to the point where they are either ingested or inhaled by people.

Models of all the above processes are integrated into the TSPA model. The submodels and some of their linkages are illustrated in Figures 4.1-1 to 4.1-4. The TSPA model is used to predict how the integrated and interdependent processes evolve over time and space, following the emplacement of the waste packages and drip shields and the ultimate closure of the potential repository.

Each of the component models included in the TSPA model directly incorporates the uncertainty in the understanding of the underlying processes, or bounds that uncertainty appropriately by selecting parameters that maximize the potential consequences of the model from an overall performance perspective (i.e., that maximize the expected dose to a human receptor). Because the component models include conservative assumptions as a method of dealing with uncertainties, the TSPA results are biased toward higher doses. The extent of that bias cannot be quantified without additional information. The uncertainties in the models are discussed in Section 3 and in the PMRs and AMRs.

In Section 4.1.1, the TSPA results for annual dose to the receptor over a 100,000-year period are presented. (See Section 3.9.1 for an explanation of the receptor, and Section 3.9.2.4 for an explanation of the method used for calculating dose TEDE.) In Section 4.1.2, a number of subsystem results are presented for the same time period, in order to provide a greater understanding of the way the system works as it degrades over time. In Section 4.1.3, the TSPA results are extended to a 1-million-year period, which includes the time when the annual dose reaches its highest values. In Section 4.1.4, TSPA results are shown for simulations with different numbers of realizations, in order to confirm that the base case includes enough realizations that the results are stable. Lastly, in Section 4.1.5, the nominal base-case results are presented for comparison with groundwater-protection requirements.

4.1.1 Overall Results for One Hundred Thousand Years

Figure 4.1-5 presents the TSPA results for the nominal scenario. The time period of regulatory interest specified in the three applicable DOE, NRC, and EPA regulations is 10,000 years after the closure of the potential repository (see Section 1.3.1.4), and the figure shows that the calculated doses are zero throughout this period. The simulations were continued out to 100,000 years to evaluate the behavior of the system after the containment of the engineered barriers is significantly degraded and to show that doses remain below the proposed limits well past 10,000 years. The 100,000-year period is not necessarily the "period of geologic stability" required to be addressed in EIS by the proposed EPA standard. Note that results for a 1-million-year period are discussed in Section 4.1.3.

The upper graph in Figure 4.1-5 shows 300 simulated dose histories along with some statistical measures of the dose distribution. The figure illustrates the temporal evolution of dose to the receptor and the uncertainty associated with that dose projection. The mean curve is generated by averaging the 300 dose values at each time step, and the percentile curves are generated by determining the location of the given percentile at each time step (for example, the median curve is generated by determining the dose that has half of the calculated doses above it and half below it at each time step). The figure illustrates that there is a considerable amount of variability in the projections of dose, but no dose occurs for any of the 300 realizations for the nominal scenario until approximately 10,000 years after closure. The differences in results for different realizations are caused by the differences in their input parameters. It is shown in Section 5.1 that the variance in dose results seen in Figure 4.1-5 is largely related to uncertainties in waste package degradation processes. At 45,000 years after closure, nearly 50 percent of the predicted doses are essentially zero, over 90 percent of the doses are less than 1 mrem/yr, and the mean dose is approximately 0.25 mrem/yr. At 100,000 years, the mean dose is approximately 70 mrem/yr, the median dose (50 percent probability) is about 10 mrem/yr, and there is about a 20 percent probability of the predicted dose being less than 1 mrem/yr.

The total dose illustrated in Figure 4.1-5 is the sum of the doses attributed to each radionuclide in the groundwater at the point of use. Figures 4.1-5 to 4.1-7 illustrate the contribution of individual radionuclides to the total dose. The lower graph in Figure 4.1-5 shows the mean dose histories for the most important radionuclides and Figure 4.1-6 shows pie charts for the mean contribution of radionuclides to the total dose at four times. These figures indicate that, initially, the dominant dose contributors are the more mobile radionuclides, ^{99}Tc and ^{129}I. As time progresses, less-mobile radionuclides, which have lower solubility and slower transport through the UZ and SZ, become the dominant dose contributors. After about 60,000 years, the dose is dominated by ^{237}Np and colloidally-transported ^{239}Pu. For completeness, Figure 4.1-7 shows the mean dose histories for the rest of the radionuclides that were tracked in the TSPA simulation. These radionuclides make very small contributions to the total dose.

4.1.2 Subsystem Results for One Hundred Thousand Years

The preceding figures describe the total system results in terms of the total annual dose to the receptor. However, it is also important to understand the causal relationships that determine these results. This section describes the evolution of the potential repository as it relates to the performance of the overall system, including the degradation of the primary containment barrier (i.e., the waste package) and the subsequent release rates of radionuclides across different barriers.

Some of the discussion to follow refers to the environmental groups that are used in calculating waste form degradation and transport out of the EBS. As discussed in Section 3.3.2, the division into environmental groups is based on infiltration during the glacial-transition climate (0 to 3 mm/yr [0 to 0.1 in./yr], 3 to 10 mm/yr [0.1 to 0.4 in./yr], 10 to 20 mm/yr [0.4 to 0.8 in./yr], 20 to 60 mm/yr [0.8 to 2.4 in./yr], or 60+ mm/yr [2.4+ in./yr]), waste type (CSNF or co-disposal waste, which refers to waste packages that have both HLW glass waste and DSNF), and seepage state (seepage at all times, some of the time, or none of the time). There is a total of 30 environmental groups, resulting from five infiltration bins, times two waste types, times three seepage states. The division of waste packages into those 30 groups is given in Table 4.1-1. Shown are the averages over all 300 realizations; note that the number of packages in each group changes from realization to realization. There is effectively little or no difference between the groups with seepage some of the time and seepage all of the time. The packages that have seepage only some of the time typically do not seep during the early dry period (i.e., during the present-day climate), but then seep all the time later on during the wetter glacial-transition climate. Thus, during the glacial-transition climate, when waste packages start failing, waste packages with seepage some of the time or seepage all of the time generally all have seepage. It can be seen from the table that only about 13 percent of the waste packages are subjected to seepage, on average. In the model, the other 87 percent are in locations where the local percolation flux is below the local seepage threshold.

Table 4.1-1. Average Division of Waste Packages into Environmental Groups

Waste Type	Seepage State	0–3 mm/yr	3–10 mm/yr	10–20 mm/yr	20–60 mm/yr	60+ mm/yr
Commercial Spent Nuclear Fuel	No Seepage	759	928	1338	2876	653
	Seepage Some of the Time	70	101	180	405	93
	Seepage All of the Time	0	1.8	8.2	98	75
Co-Disposal Waste	No Seepage	394	480	693	1487	341
	Seepage Some of the Time	33	51	89	205	44
	Seepage All of the Time	0	0.66	4.5	50	38

NOTE: 25.4 mm = 1 in.
The TSPA model has an additional 274 CSNF waste packages with stainless-steel cladding that are not included in the table. See text for explanation.

In addition to the waste packages listed in the table, the TSPA model has 274 CSNF waste packages with stainless-steel cladding. These 274 packages are placed in a separate environmental group because they are treated differently from the packages with Zircaloy cladding (the stainless-steel cladding is assumed to be perforated when the waste package fails; see Section 3.5.4.2). Because the number of packages with stainless-steel cladding is small, they are all placed together in a single environmental group, rather than creating numerous additional groups for them. The environmental conditions for the group with stainless-steel cladding were taken from the infiltration bin with the greatest number of waste packages for each infiltration case. Thus, environmental conditions from the 0 to 3 mm/yr (0 to 0.1 in./yr) infiltration bin were used for low-infiltration realizations (17 percent of the realizations), and environmental conditions from the 20 to 60 mm/yr (0.8 to 2.4 in./yr) infiltration bin were used for medium- and high-infiltration realizations (83 percent of the realizations). (For the number of waste packages in each infiltration bin, see Figure 3.3-3 and CRWMS M&O 2000 [152204], Table 5). A seepage state was needed for the stainless-steel-cladding group as well, so environmental conditions for seepage all of the time were specified. For low-infiltration realizations, there were no seepage results available for seepage all of the time in the 0 to 3 mm/yr (0 to 0.1 in./yr) infiltration bin, so the TSPA model defaulted to no seepage for those packages.

When the numbers of packages in Table 4.1-1 are added up, and the 274 CSNF waste packages with stainless-steel cladding are included, a total of 7,860 CSNF packages and 3,910 co-disposal packages is obtained, as specified in Section 3.5.1.1. Note that the proportions of waste packages in the five infiltration bins are a reflection of the spatial variability of infiltration (see Figures 3.3-3 and 3.2-8). The proportions of waste packages in the three seepage states are a reflection of the spatial variability of seepage (see Section 3.2.4.3). The large fraction of waste packages with no seepage is caused, in part, by the inclusion in the model of flow focusing, which reduces the number of waste packages that have seepage but increases the flow rate of water onto the packages that have seepage (see Section 3.2.4.3).

Water is key to performance of the potential repository system. Without water to mobilize radionuclides and provide transport pathways, the only doses that would occur would be very small doses from the few radionuclides that can be transported in gaseous form. Conceptually, the first barrier protecting the waste from water is the portion of the UZ overlying the potential repository. Surface effects such as runoff and evapotranspiration limit the amount of water that percolates to depth (Section 3.2.2), and the emplacement-drift openings have a capillary-barrier effect that diverts much of the water flow around the drifts rather than into them (Section 3.2.4). The reduction of water flow at the surface is illustrated in Table 4.1-2, which lists the ratio of mean infiltration flux to mean precipitation flux, over the area of the potential repository, for each infiltration case and climate state. The table shows that only a small fraction of the precipitation is converted to infiltration that can percolate through the mountain and potentially contact waste. Because of the stochastic nature of the seepage abstraction, it is difficult to make a similar illustration for the reduction of the amount of flow into the drifts, but Table 4.1-1 serves as an illustration of the spatial distribution of seepage into drifts. As noted previously, only about 13 percent of the waste packages are subjected to seepage, on average. However, because of flow focusing and threshold effects, the fraction of water flow entering drifts is not necessarily the same as the fraction of waste packages with seepage.

Table 4.1-2. Ratio of Infiltration to Precipitation for the Infiltration Cases

Infiltration	Present-Day Climate	Monsoon Climate	Glacial-Transition Climate
Low infiltration	0.2%	2.4%	1.1%
Medium infiltration	2.4%	4.0%	6.1%
High infiltration	4.2%	4.8%	8.5%

Source: USGS 2000 [123650], Tables 6-10, 6-14, 6-19

Before liquid water can contact the waste package, the titanium drip shield must be breached. Figure 4.1-8 shows how the drip shields fail over time in the TSPA model. The mean cumulative-failure history (i.e., the average of the failure histories for the 300 realizations) is shown, as well as the median, 5th-percentile, and 95th-percentile failure histories. The meaning of the 5th-percentile curve is that in 5 percent of the TSPA realizations that number or fewer drip shields have failed; the other percentile curves are analogous (note that median is the same as 50th-percentile). The figure shows that drip shields do not start failing until approximately 20,000 years after emplacement. Most of the drip shields have failed by about 40,000 years, though in a small fraction of realizations most of the drip shields last for more than 100,000 years.

Figure 4.1-9 shows how the waste packages fail over time in the TSPA model. The mean failure curve rises from one waste package at about 15,000 years to about 50 percent of the waste packages by 100,000 years. However, there is a great deal of spread about the mean failure curve. At the 5-percent probability level, no waste packages fail for over 90,000 years and less than 1 percent fail by 100,000 years, whereas at the 95-percent probability level, 50 percent of the waste packages fail before 50,000 years and nearly all of them fail by 100,000 years. The "waves" visible in the curves in Figure 4.1-9 result from the time discretization of the waste-package failure model (see Section 3.4). While the waste packages are generally much longer-lived that the drip shields (compare Figures 4.1-8 and 4.1-9), the first waste-package failures occur before the first drip-shield failures. This occurrence underscores the fact that the waste-package failure model is independent of dripping conditions, and therefore has little or no dependence on the presence of the drip shield (see Section 3.4).

Once the outer, corrosion-resistant barrier (Alloy-22) of a waste package has degraded, moisture can enter the package. The TSPA model conservatively neglects any containment potentially provided by the inner, less corrosion-resistant barrier of the waste package (stainless steel). In addition, no credit is taken for the pour canisters that encapsulate the HLW glass. However, corrosion-resistant Zircaloy cladding provides additional protection to about 99 percent of the CSNF. The progression of initial cladding failure is illustrated in Figure 4.1-10, which shows the mean fraction of cladding failed over time for the 20 to 60 mm/yr (0.8 to 2.4 in./yr) infiltration bin. Cladding failure only applies to the CSNF waste type. Approximately 8 percent of the cladding fails at early times when temperatures are high, mostly by creep rupture. Failure after that is caused primarily by localized corrosion, which only occurs in the presence of seepage (see Section 3.5.4.2). Note that the cladding is protected from seepage as long as either drip shield or waste package is intact; thus, the number of fuel rods with perforated cladding starts rising after about 40,000 years—after most drip shields have failed and large numbers of waste packages have started failing. For perspective, Table 4.1-1 shows that, in the TSPA model, most waste packages are in locations that have no seepage; the "never dripping" curve in Figure 4.1-10 accounts for 85 percent of the waste packages in the 20 to 60 mm/yr (0.8 to 2.4 in./yr) infiltration bin. Thus, the additional cladding failures in the "always dripping" and "sometimes dripping" groups represent only a small increase in failed cladding overall within 100,000 years.

Having examined the failure history of the primary engineered barriers, next the movement of radionuclides through the system will be presented. Figure 4.1-11 shows the release rates of three important radionuclides as they pass some of the key engineered and natural barriers. Rate of release of radioactivity from the waste form, from the waste packages, from the EBS (i.e., from the emplacement drifts), from the UZ, and from the SZ (to the biosphere) are presented.

Figure 4.1-11a shows mean release histories for ^{99}Tc, which is a very mobile radionuclide. It is highly soluble and has little or no retardation due to sorption during transport. In the TSPA model, ^{99}Tc has no sorption at all in most hydrogeologic units. Only a small amount of sorption occurs in the part of the SZ path that is through alluvium. The behavior of ^{129}I is the same as that of ^{99}Tc; it also is highly soluble and has a small amount of sorption only in the alluvium. Releases of ^{99}Tc start as soon as the first waste package failures occur, just after 10,000 years. At late times, after about 60,000 years, a quasi-steady state is achieved, in which the release rates are the same throughout the system (the rate at which ^{99}Tc reaches the biosphere is the same as the rate at which it is released from the waste form). The jumps in the waste form release rate (for example at approximately 20,000 years) occur at times when there is an increase in the rate of waste package failures (see Figure 4.1-9). The separation between the waste form and waste package curves indicates a residence time within the waste packages on the order of 10,000 years. It is seen that the waste packages are important in limiting releases of radionuclides even after they are breached. In contrast, the EBS curve overlays the waste package curve, indicating little residence time in the invert below the waste packages. Some residence time in the UZ can be seen, but there is apparently little additional residence time in the SZ. The long residence time for ^{99}Tc in the waste packages occurs because the waste packages have very small openings (cracks) when they first fail (see Figure 4.1-12). In the model, transport through the EBS is primarily diffusive for ^{99}Tc (see Figure 4.1-13a), and diffusion is quite slow when only small cracks are present. As the area available for diffusion increases, the residence time in the waste packages decreases, as shown by the convergence of the waste form curve with the other curves in Figure 4.1-11a. See Section 3.6 for details of the TSPA model for radionuclide transport within the EBS.

It is worth noting here that the EBS flow and transport abstractions include a number of conservatisms (see Section 3.6.3). Those conservatisms tend to overestimate radionuclide releases from the EBS, and therefore tend to overestimate the annual dose to the receptor 20 km (12 miles) away. Conservatisms are acceptable in TSPA simulations, but it should be kept in mind that they can affect conclusions regarding the relative importance of various release mechanisms. For example, diffusive releases from the EBS are overestimated because the diffusion time within the waste package is neglected, a continuous pathway for diffusion from the waste package to the invert is assumed always to be present after waste-package failure, and the diffusion coefficient in the invert is a conservative estimate (Section 3.6.3). The result that diffusive releases of ^{99}Tc are much greater than advective releases (and the results, discussed below, that diffusive releases of some other radionuclides are greater than or approximately equal to their advective releases) could be caused by these conservatisms, and therefore might not be realistic.

Figure 4.1-11b shows mean release histories for ^{237}Np, which is a moderately mobile radionuclide. It is more soluble than most species, though much less soluble than ^{99}Tc. It is weakly sorbing during transport. Releases of ^{237}Np from the waste form start as soon as the first waste package failures occur, but there is a lag of a few thousand years before releases reach the biosphere. The peaks in some of the curves (especially the waste package and EBS curves) result from failure of individual waste packages, or small groups of packages that fail at approximately the same time. The gap between the waste form and waste package curves is even larger than it is for ^{99}Tc, indicating an even longer residence time within the waste packages after being released from the waste form. Once again, the EBS curve overlays the waste package curve, indicating little residence time in the invert below the waste packages. Unlike the case for ^{99}Tc, the gap between the UZ and SZ curves is larger than the gap between the EBS and UZ curves, indicating a greater residence time in the SZ than the UZ. The greater residence time occurs because there is more sorption of neptunium in the SZ (especially in the alluvium) than in the UZ; however, the residence time in the SZ is very small compared to that in the waste packages. The greater residence time in the waste packages is a result of the lower solubility of neptunium as compared to technetium. The concentration of ^{237}Np cannot rise as high, which means that the concentration gradient cannot be as large, and therefore diffusive releases cannot be as large. Lower concentration reduces advective releases as well, since advective releases are proportional to source concentration. Figure 4.1-13b shows the split between diffusive and advective releases from the EBS for ^{237}Np. Unlike for ^{99}Tc, diffusive and advective releases are approximately equal for ^{237}Np after about 40,000 years. Before about 20,000 years releases are purely diffusive because the drip shields are all still intact (see Figure 4.1-8).

Advective releases are small compared to diffusive releases until about 40,000 years, when the first patch openings in the waste packages occur. Prior to that time, the waste package openings are small cracks (Figure 4.1-12), which prevent most water from entering the waste packages. In the TSPA model, the amount of seepage entering the drift above a waste package is reduced at the drip shield according to the fraction of the drip shield area that is open. The water flow is further reduced at the waste package according to the fraction of the waste package area that is open. The model for flow of water within the drifts is discussed in Section 3.6.2.1. The reduction of seepage at the drip shield and waste package is illustrated in Figure 4.1-14, which shows the mean amount of seepage into drifts for a waste package that has seepage, along with the mean amount of flow through the drip shield and into the waste package. CSNF packages with seepage some of the time in the 20 to 60 mm/yr (0.8 to 2.4 in./yr) infiltration bin were chosen for the illustration because that is the largest group with seepage (see Table 4.1-1). The sharp rise in water entering the waste package at approximately 40,000 years (Figure 4.1-14) explains the sharp rise in advective releases from the EBS at the same time (Figure 4.1-13b).

Figures 4.1-11c and 4.1-11d show mean release histories for ^{239}Pu. It is less soluble than ^{237}Np. When transporting as a solute it is highly sorbing, so ^{239}Pu can only transport significantly if facilitated by colloids (the same is true for all highly sorbing species, including americium and other isotopes of plutonium). Two types of colloid-facilitated transport are modeled: reversible (Figure 4.1-11c) and irreversible (Figure 4.1-11d). "Reversible" refers to radionuclides attached to colloids by reversible sorption interactions, while "irreversible" refers to radionuclides that are permanently attached to colloids (the radionuclides are actually part of the structure of the colloids). More information on modeling of colloidal transport in the EBS, the UZ, and the SZ can be found in Sections 3.6, 3.7, and 3.8, respectively.

Figure 4.1-11c shows that releases of ^{239}Pu from the waste form start as soon as the first waste package failures occur, but there is a lag of several thousand years before releases of ^{239}Pu reversibly attached to colloids reach the biosphere. The very large gap between the waste form and waste package curves indicates a long residence time of reversible colloids within the waste packages after being released from the waste form. The reason for the long residence in the waste package is the same as discussed for ^{237}Np, and the residence time is even longer for ^{239}Pu because the solubility limit for plutonium is lower than that for neptunium. The EBS curve overlays the waste package curve after about 30,000 years, once again indicating little residence time in the invert below the waste packages. A significant residence time in the UZ is indicated by the gap between the EBS and UZ curves, and several thousand years in the SZ as well. The EBS curve is anomalous before 30,000 years, in that releases from the EBS are higher than releases from the waste package. This occurrence results from ingrowth of ^{239}Pu from ^{243}Am, which is released from the waste packages in greater quantities than ^{239}Pu at early times, and then decays and is released from the EBS as ^{239}Pu.

Figure 4.1-11d shows mean release histories for ^{239}Pu irreversibly attached to colloids. Comparison to Figure 4.1-11c shows that releases of irreversible colloids are much lower than releases of reversible colloids in the TSPA model. The EBS and UZ curves both overlay the waste package curve, indicating a short residence time for irreversible colloids in the EBS and the UZ. There is a small gap between the UZ and SZ curves, indicating some residence time in the SZ, but only a few thousand years. The longer residence in the SZ results from inclusion of a retardation factor for irreversible colloids in the SZ transport model (see Section 3.8.2.4), whereas such retardation is neglected in UZ and EBS transport. A very long residence time in the waste packages is indicated. This long residence time results because there is very little diffusive release of colloids, (because the diffusion coefficients of colloids are much lower than the diffusion coefficients of the dissolved species; see Section 3.6.2.2) so significant releases do not occur until the water flow into waste packages becomes significant, at about 40,000 years (see Figure 4.1-14).

Mean diffusive and advective releases from the EBS for reversible and irreversible ^{239}Pu are shown in Figures 4.1-13c and 4.1-13d, respectively. Figure 4.1-13d shows that diffusive releases of irreversible colloids are very low, as just discussed. In contrast, Figure 4.1-13c shows that there are significant diffusive releases of reversible colloids from the EBS. The reason for this is that the reversible component also includes ^{239}Pu that transports as a solute. Because of its high sorption, normally solute transport of ^{239}Pu is negligible, but sorption of all species was conservatively neglected for EBS transport, thus allowing significant releases from the EBS of dissolved ^{239}Pu. This dissolved ^{239}Pu then continues its transport through the UZ and SZ reversibly attached to colloids. Figures 4.1-13c and 4.1-13d also show that there is a small amount of advective release of the reversible component from the EBS prior to 40,000 years, whereas advective releases of the irreversible component do not begin until after 40,000 years. The earlier advective releases of reversible ^{239}Pu can also be attributed to transport of solute within the EBS.

Figure 4.1-15 shows releases from the EBS split into the mean releases from CSNF (top) and from co-disposal waste (bottom). Even though co-disposal waste accounts for only approximately 10 percent of the total waste mass, the releases of ^{99}Tc and ^{239}Pu from co-disposal waste and CSNF are comparable. The CSNF releases are relatively low because of the additional cladding barrier that the CSNF has (see Figure 4.1-10). An exception is ^{237}Np, which has co-disposal releases much lower than CSNF releases. This difference results from higher pH in the co-disposal packages, which lowers the in-package solubility of neptunium significantly (see Section 3.5.5.1). Technetium and plutonium solubilities are not sensitive to pH, so they are not affected in the same way. Another difference between CSNF and co-disposal waste is that the irreversible colloids come only from the latter. They are essentially tiny pieces of HLW glass that break off as the glass waste is degrading.

In the TSPA model, EBS releases are transferred to the UZ transport model separately for the five infiltration bins so that the releases in wetter repository regions are correlated with transport through wetter parts of the UZ (see Section 3.7.2). The mean EBS releases for the five infiltration bins are shown in Figure 4.1-16 for ^{99}Tc (top) and ^{237}Np (bottom). The differences in the two figures are a further illustration of the fact that ^{99}Tc releases are dominated by diffusion, whereas ^{237}Np releases have a significant advective component. In Figure 4.1-16, for ^{99}Tc, the curves are ordered simply by the number of waste packages in each bin. From Table 4.1-1 it can be seen that the 20 to 60 mm/yr (0.8 to 2.4 in./yr) bin has the greatest number of waste packages, with over twice as many packages as the next largest bin. The next largest bin is the 10 to 20 mm/yr (0.4 to 0.8 in./yr) bin, then 3 to 10 mm/yr (0.1 to 0.4 in./yr), 0 to 3 mm/yr (0 to 0.1 in./yr), and lastly 60+ mm/yr (2.4+ in./yr). The releases are closely proportional to the number of waste packages, because the diffusive releases are similar for all waste packages. In particular, the diffusive releases have little if any dependence on the local UZ flow. In contrast, in the bottom graph of Figure 4.1-16, for ^{237}Np, the order of the curves reflects the amount of water flow: The 20 to 60 mm/yr (0.8 to 2.4 in./yr) bin is still highest since it has so many waste packages, but the 60+ mm/yr (2.4+ in./yr) bin is second highest even though it has the smallest number of waste packages. The 10 to 20 mm/yr (0.4 to 0.8 in./yr) bin is next, then 3 to 10 mm/yr (0.1 to 0.4 in./yr), and lastly 0 to 3 mm/yr (0 to 0.1 in./yr).

Radionuclides reaching the water table are separated into four regions for input to the SZ transport model (see Section 3.8.2.2). The mean UZ releases for the four regions are shown in Figure 4.1-17 for ^{99}Tc (top) and ^{239}Pu (bottom). The curves in the top graph of Figure 4.1-17, for ^{99}Tc, match the general picture of UZ transport described in Sections 3.2.3.4 and 3.7.4. In this picture, there is significant lateral flow and transport to the east (down-dip) and then down in the northern part of the potential repository because of the presence of perched water below the potential repository, whereas in the southern part of the potential repository flow and transport are more nearly vertical, with much less lateral movement. As shown in Figure 3.8-14, Regions 1 and 3 are directly below the potential repository, with Region 1 in the north and Region 3 in the south; Regions 2 and 4 are to the east of the potential repository area (Region 2 in the north and Region 4 in the south), where laterally diverted radionuclides would reach the water table. In the top graph of Figure 4.1-17, the highest UZ releases are in Regions 3 and 2, representing transport directly down to the water table below the southern part of the potential repository and laterally to the east and down in the northern part of the potential repository. UZ releases from Regions 1 and 4, representing transport directly down through the perched water in the northern part of the potential repository and laterally to the east and down in the southern part of the potential repository, are lower, confirming that those pathways are less important in the model.

The bottom graph of Figure 4.1-17, for ^{239}Pu, is included because of some interesting differences. Compared to the curves for ^{99}Tc, the ^{239}Pu releases in the southern part of the potential repository are relatively lower (that is, curves 3 and 4 for ^{239}Pu are lower relative to curves 1 and 2 than they are for ^{99}Tc). The lower (or slower) ^{239}Pu releases in the south are a result of the greater occurrence of flow and transport through the porous rock matrix in the south, which is a result of the greater occurrence of vitric Calico Hills nonwelded tuff in the south. Plutonium is highly sorbing on the rock matrix, whereas sorption on fracture faces is neglected in the TSPA model. The ^{239}Pu transport is primarily via reversible sorption on colloids, and when the transport is through a highly sorbing porous medium some of the radionuclides sorb to the porous matrix instead of the colloids, which causes an effective retardation of the transport.

Figure 4.1-18 shows the mean releases of ^{99}Tc from the SZ for the four source regions (the same four regions discussed above). Shown are the release rates at 20 km (12 miles) distance, where the radionuclides are assumed to be pumped up a well and used for farming and household activities. Releases from the four regions are added together to obtain the total release rate, which is then divided by the groundwater-usage volume and multiplied by BDCF for each radionuclide to calculate the dose to the receptor (see Section 3.9.2.4). The final dose curve for ^{99}Tc is shown in Figure 4.1-5. Comparison of Figure 4.1-18 (SZ releases) with the top graph of Figure 4.1-17 (UZ releases) shows no significant difference between them, thus indicating no significant difference in the SZ transport of radionuclides that start in the different source regions. Though not shown, transport of ^{237}Np and ^{239}Pu also shows no significant dependence on the source location.

4.1.3 Results for One Million Years

The bulk of the analyses presented in this and other sections of this document have the objective of evaluating the ability of the Yucca Mountain site and its related natural barriers and the engineered barriers associated with the Site Recommendation design to meet performance objectives specified in the applicable environmental standard set by the U.S. EPA in the proposed 40 CFR Part 197 (64 FR 46976 [105065]) and the licensing criteria specified in the U.S. NRC's proposed 10 CFR Part 63 (64 FR 8640 [101680]). Both of these regulations at present require performance assessments to extend through the compliance period of 10,000 years. The analyses conducted to evaluate the ability of the site to meet these standards during the compliance period have been extended to longer time periods, specifically to 100,000 years, in order to assure that no dramatic degradation of the performance occurs after the 10,000-year compliance period. Additional analyses of peak dose to 1,000,000 years have also been conducted.

Both the compliance period models and analyses and the extension of these models and analyses to 100,000 years have used the principal of bounding the uncertainties in order to assure that the projected dose would be an overestimation of the range of possible performance. Extending the analyses significantly beyond the 10,000-year compliance period, especially when using the models and analyses purposely designed to be bounding, raises questions about the interpretation of the results. As noted by EPA in their evaluation of the potential for compliance periods of greater than 10,000 years: "as the compliance period is extended to such lengths, uncertainty increases and the resulting projected doses are increasingly meaningless from a policy perspective" (40 CFR Part 197 [64 FR 46976 [105065], p. 46994]).

In addition to the lack of policy meaning for such long term projections, EPA has noted that there is no policy basis for determining "the level of proof or confidence necessary to determine compliance based upon projections of hundreds-of-thousands of years into the future" (40 CFR Part 197 [64 FR 46976 [105065], p. 46995]). In addition, the EPA notes that although the "far future can be bounded, a large and cumulative amount of uncertainty is associated with those numerical projections." (40 CFR Part 197 [64 FR 46976 [105065], p. 46995]).

In the evaluations of potential compliance with the above regulatory requirements, overall conservative (i.e., pessimistic) calculations of the potential impacts of this system to estimate the potential risks of the repository for 10,000 years were used. Much effort has gone into providing an appropriately defensible technical basis for the 10,000-year compliance period analyses recognizing the uncertainty in these projections. To gain insight into the 10,000-year performance results, calculations have been carried out to 100,000 years. These were calculations expressly done to gain insight into the robustness of the regulatory compliance-period case, however. It is not a good indicator of expected performance to take conservative calculations for 10,000 years and project them significantly beyond their basis.

Much of the 10,000-year calculational structure and basis properly supports a longer term calculation, but there are some features and processes that are not important to 10,000-year performance that can therefore be bounded or even neglected. However, these same features or processes become important in the longer time frames if the goal is to project a reasonable expectation of performance for the longer term. Some purposely neglected or bounded features and processes that become potentially important to performance only after the time period of regulatory concern would act to increase potential risk, others to decrease potential risk. In order to provide a reasonable expectation of a peak dose beyond 10,000 years, however, all potentially important processes must be appropriately included.

With these qualifiers in mind, three separate total system analyses of peak dose have been performed out to 1,000,000 years. In the first case, the models developed for the 10,000-year compliance period have simply been extrapolated out to the peak dose. The result of utilizing this conservative assumption is illustrated in Figure 4.1-19a. The key radionuclides affecting the dose assessment are illustrated in Figure 4.1-19b. In examining these results, several observations are possible, including:

- The peak mean dose is about 490 mrem/yr and occurs at about 270,000 years

- The dominant radionuclide contributing to the peak dose is ^{237}Np, with lesser contributions provided by colloids of thorium, radium, and plutonium

- The distribution of peak dose indicated by the range of results for the different realizations spans several orders of magnitude. This range is significantly less than the range of possible dose projections at earlier times because the uncertainty in the performance of some key aspects of the engineered and natural barriers decreases with time. In particular, after about 100,000 years, all of the waste packages and drip shields have been sufficiently degraded to allow the waste contained in these packages to be available for mobilization and transport.

In the second case, an alternative representation for the solubility of actinide elements (e.g., neptunium, plutonium, and americium, in particular) was utilized that employed the observation that these radioisotopes are commonly entrapped in secondary uranium phases when waste fuel alters. These secondary phases induce a significantly lower solubility of these key radionuclides as discussed in Section 3.5.5.3. Employing these expected solubilities for the expected dose projection results in significantly lower doses as illustrated in Figure 4.1-20. The key radionuclides contributing to these dose estimates are indicated on the bottom plot in Figure 4.1-20. The following observations assist in explaining the differences between the two analyses:

- The peak mean dose decreases from 490 mrem/yr to about 30 mrem/yr at 1,000,000 years.

- The dominant radionuclides contributing to the peak dose are the more soluble radionuclides such as ^{99}Tc and the colloidally transported species of ^{239}Pu and ^{242}Pu. The dose attributed to ^{237}Np is reduced by several orders of magnitude from the previous case, reflecting the significantly reduced soluble fraction when long-term secondary phases are included in the analysis.

In the third case, the long-term climate was adjusted to account for the expectation that glacial climate periods are likely to recur several times during the 1,000,000-year simulation period. As described in Section 3.2.5, these glacial climate periods have the effect of increasing the net infiltration (due to increased precipitation and reduced temperature) which in turn causes an increase in percolation flux and seepage at the repository horizon. In addition, the glacial climates have the effect of raising the water table and increasing the advective flux in the saturated zone. The result of utilizing the modified long-term climate in conjunction with the expected reduction in the long term solubilities due to the formation of secondary phases is illustrated in Figure 4.1-21. The following observations assist in explaining the effects of these changes on the projected mean peak dose:

- The peak mean dose is about 120 mrem/yr at about 700,000 years.

- The timing of the mean peak dose is dependent on the timing and magnitude of the change in the net infiltration rate. This correlation is primarily driven by the fact that increased infiltration yields an increase in seepage, which in turn yields an increase in the advective release of solubility-limited and colloidally transported radionuclides from the engineered barrier system.

- The "spikiness" of the mean dose response when the post-10,000-year climate changes are implemented is a result of the relative abruptness in the timing of the climate change and the rapid propagation of the induced changes through the system as well as the relatively short duration of the full-glacial climate states. The full-glacial climate states have durations on the order of 10,000 years, and once the climate returns to intermediate conditions, the infiltration and seepage return to the base-case conditions, including the base-case release and dose projections.

In summary, the peak of the expected dose curve is a function of the degree of conservatism incorporated in the models and analyses used to produce the peak dose estimate. Because the base-case models used in the development of the nominal performance projections were designed to be reasonably conservative to maximize their defensibility during the 10,000-year compliance period, they are less appropriate for projections of the peak dose. More appropriate representations would include considerations of the long-term (post-10,000-year) climate states and the long term effects of secondary phases. Although these alternative representations are less appropriate for the compliance period analyses, they are more suitable for the peak dose assessments.

In any assessment of the peak dose, the cautions provided by EPA should be considered in the interpretation of the results, namely "extremely long-term calculations are useful only as indicators, rather than accurate predictors, of the long-term performance of the Yucca Mountain disposal system" (40 CFR Part 197 [64 FR 46976 [105065], p. 46996]).

4.1.4 Precision of Probabilistic Results

The number of realizations to use in a Monte Carlo simulation is a significant issue in terms of reliable analyses and proper allocation of computing resources. All probabilistic results presented thus far were based on 300 realizations for developing estimates of the mean annual dose and its corresponding uncertainty. In this section, it is shown that 300 realizations are adequate to calculate mean annual doses reliably. To this end, the TSPA model was re-run using 500 realizations, and the results compared to those obtained for the base-case simulation with 300 realizations.

Figure 4.1-22 shows a comparison of the statistics of the dose history for the two cases. Results of a simulation with 100 realizations are also included in the comparison. The 100-realization mean dose history is used for comparison with a number of sensitivity cases in Sections 5.2 and 5.3. Figure 4.1-22 indicates very good agreement over the 100,000-year time period for the mean, median, 5th percentile, and 95th percentile dose histories. These results verify that various measures of uncertainty in the projected dose, quantifying the central tendency as well as the overall spread, are essentially equivalent for the 500- and 300-realization simulations. Even the 100-realization simulation appears robust enough to predict the mean of the annual-dose distribution through time. However, a sample size of 100 is less reliable for estimating the tails of the annual-dose distribution or the dependency of the uncertain annual-dose distribution on the uncertain distributions of the various input parameters (see Section 5.1). Thus, the 300-realization sample size is considered to be an appropriate compromise for analysis of the base case, although the 100-realization sample is considered to be adequate for comparing the trends in predictions of mean annual dose for the various sensitivity cases in Sections 5.2 and 5.3.

4.1.5 Groundwater Protection

An analysis of groundwater protection was conducted in accordance with the EPA's proposed 40 CFR 197.35 (64 FR 46976 [105065]). The proposed rule is based on meeting three maximum levels. The first is that the maximum concentration of ^{226}Ra and ^{228}Ra shall not exceed 5 pCi/L in a representative volume of groundwater at the point of compliance, considering both natural sources and releases from the potential repository. The second limit is that the gross alpha activity (excluding radon and uranium) shall not exceed 15 pCi/L in the representative volume of groundwater at the point of compliance, again from both natural sources and releases from the potential repository. The third limit is that beta- and photon-emitting radionuclides released from the potential repository shall not cause a dose in excess of 4 mrem/yr to the whole body or any organ of a human receptor.

The proposed rule specifies that the representative volume of groundwater would be withdrawn annually from an aquifer containing less than 10,000 mg/L of total dissolved solids, be centered on the highest concentration in the plume of contamination at the point of compliance, and contain 1,285 acre-ft (1,591,000 m^3) of water. For the analysis presented here, the point of compliance is taken to be 20 km (12 miles) from the potential repository, near the intersection of U.S. Route 95 and Nevada State Route 373 (alternative 2 for proposed 40 CFR 197.37 [64 FR 46976 [105065]). There is a Nevada Department of Transportation well near that highway intersection, and water from that well has been measured to have 385 mg/L of total dissolved solids (DTN: GS971000012847.004 [149980]), satisfying the first part of the definition of the representative volume. In this analysis, all radionuclides that reach a distance of 20 km (12 miles) from the potential repository in any given annual period are contained in 1,285 acre-ft of water to determine the concentration. Taking all radionuclides in this manner produces the highest estimate of concentration that is possible for the specified volume of water. The exact location or dimensions for the representative volume could be different in each Monte Carlo realization with this method. The proposed rule also specifies a time period of compliance of 10,000 years. There are no releases from the potential repository in 10,000 years, but results are presented for 100,000 years to show that they remain below the proposed limits well past 10,000 years.

To address the proposed rule, the existing background concentrations of ^{226}Ra, ^{228}Ra, and radionuclides contributing to gross alpha activity (excluding radon and uranium) must be known. Gross alpha activity (excluding radon and uranium) was measured at the Nevada Department of Transportation well and reported as 0.4 ± 0.7 pCi/L (CRWMS M&O 1999 [150420], Section 3.2.1). The same reference reports measurements at other wells and springs in the vicinity of Yucca Mountain and Amargosa Valley ranging from –0.2 ± 0.5 to 2.7 ± 3.0 pCi/L. Concentrations of ^{226}Ra and ^{228}Ra were not reported in that reference because the gross alpha activity was below 5 pCi/L. However, another source reports ^{226}Ra concentration of 0.04 pCi/L and ^{228}Ra concentrations as less than 1 pCi/L at the Nevada Department of Transportation well (DTN: GS971000012847.004 [149980]). The measurement errors are not reported in that reference. Measurements in the same reference at other wells and springs in the vicinity of Yucca Mountain and Amargosa Valley range from 0.03 to 0.5 pCi/L for ^{226}Ra and from less than 1 to 1.1 pCi/L for ^{228}Ra.

Figure 4.1-23 shows the calculated activity concentration of ^{226}Ra and ^{228}Ra, not including the background concentration. These isotopes have short half-lives and are created by ingrowth from other (parent) radionuclides released from the potential repository. Their transport is not modeled explicitly, but rather by assuming secular equilibrium with their parents, and thus these concentrations could be overestimates of the actual values. The sum of the two radium concentrations is shown in Figure 4.1-24, and is virtually identical to the concentration history of ^{226}Ra. Note that the concentrations of both isotopes are zero over 10,000 years, and are significantly less than the 5-pCi/L limit over 100,000 years. The highest concentration shown is approximately 0.06 pCi/L. Given this result, the activity concentration of ^{226}Ra and ^{228}Ra with background included is still less than 5 pCi/L over the whole 100,000-year period. From above, the background concentration of ^{226}Ra plus ^{228}Ra is less than 1.04 pCi/L at the point of compliance. Adding the releases from the repository brings the total up to 1.1 pCi/L at most, within 100,000 years.

In addition to the summed ^{226}Ra and ^{228}Ra activity, Figure 4.1-24 shows the calculated gross alpha activity (excluding radon and uranium) over 100,000 years, not including the background activity. Radionuclides included in this calculation are ^{210}Pb, ^{226}Ra, ^{228}Ra, ^{227}Ac, ^{229}Th, ^{230}Th, ^{231}Pa, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, and ^{243}Am. The activity concentrations of these radionuclides are zero over 10,000 years. The highest concentration shown is approximately 14 pCi/L at 100,000 years. If the background gross alpha activity of 0.4 ± 0.7 pCi/L at the point of compliance is added to the calculated gross alpha activity, the sum approaches the limit of 15 pCi/L at 100,000 years. However, the limit only applies for 10,000 years.

Figure 4.1-25 shows the contribution of radionuclides to the calculated gross alpha activity. The major contributors are ^{237}Np and ^{239}Pu. It is expected that ^{237}Np will continue to be the major contributor over later times, because its half-life is 2.1 million years, while ^{239}Pu should decrease in significance over later times because its half-life is 24,000 years. Radionuclides whose abundance is mainly controlled by ingrowth, e.g., the $^{230}Th/^{226}Ra/^{210}Pb$ group and ^{229}Th, show increasing activity at 100,000 years. Two groupings of radionuclides are covered by the same line type in the legend: the $^{230}Th/^{226}Ra/^{210}Pb$ group and the $^{231}Pa/^{227}Ac$ group. These are radionuclides whose transport is modeled by assuming secular equilibrium, and therefore they have the same activity.

Figure 4.1-26 shows the concentrations of the three beta- and photon-emitting radionuclides that are important to TSPA-SR and the proposed EPA rule: ^{14}C, ^{99}Tc, and ^{129}I. For ^{14}C, the critical organ is fat and the concentration in drinking water that can cause a 4 mrem/yr dose is 2000 pCi/L. For ^{99}Tc, the critical organ is the gastrointestinal tract and the concentration in drinking water that can cause a 4 mrem/yr dose is 900 pCi/L. For ^{129}I, the critical organ is the thyroid and the concentration in drinking water that can cause a 4 mrem/yr dose is 1 pCi/L. (These concentrations are based on 2 L/day drinking-water intake and are taken from U.S. Environmental Protection Agency 1976 [151667], Table IV-2A.) As there are no radionuclides with the same critical organ among these three, and none involve the total body, the concentrations in Figure 4.1-26 can be compared directly with the given concentrations. First note that the concentrations for all three radionuclides are zero for 10,000 years. The maximum concentration of ^{14}C occurs at about 24,000 years and is approximately 0.07 pCi/L. The concentration decreases after 24,000 years because of the relatively short ^{14}C half-life of 5,700 years. The maximum concentration of ^{99}Tc occurs at 100,000 years and is approximately 640 pCi/L. The maximum concentration of ^{129}I occurs at 100,000 years and is approximately 1.7 pCi/L; the concentration of ^{129}I exceeds 1 pCi/L at about 65,000 years.

The dose to a critical organ from a radionuclide can be estimated by multiplying the calculated concentration of the radionuclide in groundwater by the ratio of 4 mrem/yr over the respective concentration that would cause a 4-mrem/yr dose.. For example, the dose to fat from ^{14}C can be estimated by multiplying the groundwater concentration of ^{14}C (in pCi/L) by 4 mrem/yr and dividing by 2000 pCi/L. Figure 4.1-27 shows the doses caused by ^{14}C, ^{99}Tc, and ^{129}I to their respective critical organs when estimated in this manner. Again, there is no dose in the first 10,000-yr period. At 100,000 years, a dose of approximately 3 mrem/yr to the gastrointestinal tract is estimated from ^{99}Tc, a dose of approximately 7 mrem/yr to the thyroid is estimated from ^{129}I, and a relatively insignificant dose to fat is estimated from ^{14}C.

Again, remember that many of the component TSPA models include conservative assumptions as a method of dealing with uncertainties, and as a result the TSPA results are biased toward higher doses.

4.2 TOTAL SYSTEM PERFORMANCE FOR THE DISRUPTIVE SCENARIO CLASS

As described in Section 3.10, igneous activity has been identified as the primary disruptive event that has a potential to affect long-term performance of the potential repository. The FEPs relevant to igneous activity that have been retained following the screening process (see Section 3.10.2.1) have been included in the TSPA-SR through two separate models for igneous disruption: a model for volcanic eruptions that intersect drifts and bring waste to the surface, and a model for igneous intrusions that damage waste packages and expose radionuclides for groundwater to transport. The TSPA-SR models and parameter values used to analyze radionuclide releases and possible doses to the exposed individual are described in detail in Section 3.10. Proposed regulatory requirements (see Section 1.3) require evaluations of disruptive scenarios for 10,000 years following repository closure. To provide additional confidence in the adequacy of the 10,000-year analysis, results of the TSPA-SR analysis are described in this section for igneous disruptions during the first 50,000 years after repository closure. Unlike the results for the nominal performance scenario class, described in Section 4.1, analyses of igneous disruption have not been extended to 100,000 years because of the additional computational burden imposed by the need to evaluate events at different times. Results shown in this section are suitable, however, for evaluating system-level performance during the period when expected annual dose is dominated by igneous disruption.

4.2.1 Incorporating Event Probability in the Total System Performance Assessment-Site Recommendation

The approach taken in the TSPA-SR to calculating doses resulting from igneous disruption of the repository is consistent with the probabilistic methodology described in NRC guidance (see NRC 2000 [149372], Section 4.4), where the NRC describes the method for weighting consequences of individual scenarios by their probability before summing the scenario consequences to determine the overall expected annual dose. (See also Section 3.4 of Reamer [1999 [119693]] for a discussion of the NRC's implementation of this approach for igneous activity.) Scenario consequences are multiplied (weighted) by the probability of the occurrence to yield an appropriate estimate of the overall risk posed by low-probability events. As described in Section 3.10.1, the probability of intrusive igneous disruption is extremely low (the mean annual probability is 1.6×10^{-8}), and the annual probability of more than one igneous disruption occurring during the period of repository performance is far less. The TSPA-SR therefore considers only a single igneous disruption of the repository during the next 50,000 years, occurring with a mean 50,000-year probability of approximately 8×10^{-3} (corresponding to a 10,000-year probability of approximately 1.6×10^{-4}). The year in which the igneous disruption might occur is uncertain, and the TSPA-SR calculates consequences for disruptions at many different times, and weights the consequences by the probability that the disruption could have occurred in the time interval represented by that simulation.

4.2.1.1 Incorporating Eruptive Event Probability

For the eruptive scenario, the TSPA-SR examines the consequences of eruptions occurring in each 31.25-year time interval during the 50,000-year period and weights the consequences of each eruption by a probability equal to 31.25 times the annual probability of an eruption that intersects the repository. This probability is sampled from the distribution characterizing uncertainty in the probability of igneous intrusion (i.e., a mean annual probability of 1.6×10^{-8}), multiplied by a factor of 0.36 to account for the fraction of intrusive events for which the associated eruption does not intersect the drifts, and then multiplied by 31.25 to yield a 31.25-year eruption probability. (For example, the mean 31.25-year eruption probability is 1.8×10^{-7}). Because eruptions are considered in each 31.25-year interval, each 50,000-year realization of the TSPA-SR GoldSim model thus includes consequences of 1,600 eruptions, weighted appropriately so that the mean probability of a single eruptive event in 50,000 years is 8×10^{-3} times 0.36, or 2.88×10^{-3}. The average dose resulting from an eruptive event for a single realization is determined by calculating doses resulting from igneous events in each 31.25-year period, multiplying by the sampled probability, and adding the doses from each eruptive event.

Computationally, the analysis takes advantage of the simplifying assumption that, although the physical properties of an eruption and the response of the repository are uncertain, this uncertainty is not dependent on the time at which the event occurs. Volcanic eruptions 100 years after closure are assumed to have the same properties as volcanoes 50,000 years after closure. Because waste packages that are intersected by an eruptive conduit are assumed to be sufficiently damaged that they provide no further protection for the waste (see Section 3.10.2.2.2) regardless of the time of the event, effects of the eruption on the repository are also assumed to be constant through time. These assumptions allow the GoldSim code (Golder Associates 2000 [151202]) to use ASHPLUME results (CRWMS M&O 2000 [151349]) for a single volcano occurring at the beginning of the 50,000-year simulation to represent the behavior of similar volcanoes occurring in each 31.25-year interval. Total radionuclide mass concentrations at the receptor point are calculated by ASHPLUME (CRWMS M&O 2000 [151349]) for the first eruptive event, and are then allocated to individual radionuclides according to their relative abundance in the inventory before being used to calculate doses. Doses from eruptions occurring at later times are calculated using the same initial radionuclide concentrations, adjusted to account for radioactive decay and ingrowth.

Uncertainty is included in the analysis of eruptive consequences by completing multiple realizations using different sampled values for input parameters that characterize uncertainty in understanding the system. As described in Section 3.10, these uncertainties include parameters that describe event probability, eruptive processes, the response of the repository, and the biosphere. For the TSPA-SR, results are calculated assuming wind blows to the south in all realizations. This assumption is unrealistic, but provides a conservative bound to uncertainty regarding wind direction and compensates for uncertainty regarding the possibility that contaminated ash deposited by winds blowing in directions other than south might later reach exposed individuals.

4.2.1.2 Incorporating Intrusive Event Probability

The potential repository's response to igneous intrusion cannot be assumed to be constant through time (for example, climate change and the thermal evolution of the system result in different groundwater flux at different times), and consequences of intrusions at different times must be analyzed separately using complete 50,000-year simulations of the full groundwater transport model. This poses a heavy computational burden, and makes it impractical to simulate multiple intrusive events at each time interval, as is done for the eruptive releases. For computational efficiency, igneous intrusions are simulated using a simpler approach in which the time of intrusion is sampled randomly from the 50,000-year period, and the probability associated with each simulation is the full 50,000-year probability, with a mean value of 8×10^{-3}. Although this approach results in statistically sparser analysis of uncertainty than the approach taken for the eruptive scenario, it yields the correct overall probability of igneous intrusion during the period of interest. The approach incorporates both uncertainty in the intrusive event's probability (by sampling from the distribution of annual probabilities) and uncertainty in the system's response to disruption (by sampling uncertain parameters describing damage to the repository and subsequent radionuclide transport).

4.2.2 TSPA-SR Results for the Igneous Disruption Scenario Class

The TSPA-SR results include 5,000 realizations, each of which represents a different set of input parameters. The probability-weighted doses from both eruptive and intrusive events calculated as described in the preceding sections are added together to give the total probability-weighted dose from igneous disruption. Results of each realization are presented as a 50,000-year probability-weighted dose history that includes 1,600 eruptions occurring at 31.25 year-intervals and a single igneous intrusion occurring at a random time.

Figure 4.2-1 shows a range of probability-weighted dose histories representing possible doses to an exposed individual following disruption of the potential Yucca Mountain repository by igneous activity. Results do not include doses that might result from the nominal performance of the repository, in the absence of igneous activity. These doses are discussed in Section 4.1. Rather than showing the full set of 5,000 realizations included in the analysis, which would result in a display too dense to interpret, the figure shows 500 individual curves (in gray) that represent every tenth realization from the total of 5,000 that were completed. These curves display probability-weighted annual dose rates calculated using different sets of sampled values for uncertain input parameters in the model. The range of results shown by these individual curves displays the uncertainty in the calculated dose history resulting from uncertainty in model parameter values. Four additional curves, shown in color, provide summary information about the distribution of results from the full set of 5,000 realizations. The mean curve, shown in red, is the average probability-weighted annual dose rate and is the performance measure appropriate for comparison to the limit specified in the proposed regulation (see Section 1.3.1.4). The percentile curves, shown for the 95th, 50th (i.e., median), and 5th percentile, show an annual dose rate that is greater than 95 percent (or 50 or 5 percent) of the calculated values at that time. The mean curve lies above the 95th percentile curve throughout the interval between approximately 3000 and 8000 years because the mean is dominated by the relatively small fraction of the total number of realizations that contribute to a high groundwater dose rate at early times. The number of realizations contributing to this pathway increases through time as the cumulative probability of an intrusion having occurs increases, causing the 95th percentile curve to climb above the mean at later times. Figure 4.2-2 shows the mean probability-weighted dose histories for the individual radionuclides that contributed to the total igneous dose rates shown in Figure 4.2-1. These individual radionuclide doses are discussed in more detail in the following paragraphs, in the context of the discussion of the mean total igneous dose history.

For approximately the first 2,000 years, the dose history is a smooth curve dominated by the effects of a volcanic eruption. As shown in Figure 4.2-2, the probability-weighed mean dose during this period reaches a peak of approximately 0.004 mrem/yr roughly 300 years after repository closure, and then drops off due to radioactive decay of the relatively shorter-lived radionuclides that contribute to doses from the ashfall exposure pathway. As shown in Figure 4.2-2, the major contributors to the eruptive dose are ^{241}Am and ^{240}Pu, ^{239}Pu, and ^{238}Pu. ^{90}Sr is a significant contributor at extremely early times, but drops off rapidly because of radioactive decay. Inhalation of resuspended particulates in the ash layer is the primary exposure pathway during this period, and the smooth decline of the mean dose curve from approximately 300 to 2,000 years results from decay of ^{241}Am, which has a half-life of 432 years.

From approximately 2,000 years after closure onward, the mean igneous dose is dominated by groundwater releases from packages damaged by igneous intrusion. The irregular shape of the curve from this point forward is in part a result of the complex groundwater transport processes, and in part also reflects the occurrence of intrusive events at random times, rather than the prescribed intervals used for the extrusive simulations. Close examination of Figure 4.2-1 shows that individual realizations display distinct peaks occurring at times that are controlled by the sampled time of intrusion and the time required for radionuclide transport through the geologic system. The intrusive event may occur at any time, and the first appearance of groundwater doses in the mean curve at approximately 2,000 years reflects retardation during transport, rather than the absence of intrusions at earlier times. The observation that some of the 500 individual curves continue to be dominated by the smooth eruptive doses for essentially all of the 50,000-year period indicates either that for those realizations the sampled time of intrusion was relatively late or that, in some cases, retardation of radionuclides during transport in the geologic system was effective for a relatively long period of time.

The overall probability-weighted mean igneous dose rate reaches a peak during the first 10,000 years of approximately 0.08 mrem/yr, occurring at 10,000 years. At later times, the calculated mean igneous dose rate is higher, increasing slowly to approach 0.2 mrem/yr at the end of the 50,000-year period. This peak mean igneous dose is dominated entirely by the groundwater releases following igneous intrusion. As shown on Figure 4.2-2, ^{239}Pu and ^{237}Np are the primary contributors to the peak mean dose.

Figures 4.2-3 and 4.2-4 provide an alternative display of the relative contributions of individual radionuclides to the total probability-weighted igneous dose rate at 1,000 and 10,000 years following repository closure. At 1,000 years after closure (Figure 4.2-3), the igneous dose rate is dominated completely by the eruptive releases, and ^{241}Am contributes more than half of the total. ^{241}Am and ^{240}Pu together account for more than three-quarters of the total, and essentially all of the dose comes from ^{241}Am, ^{240}Pu, ^{239}Pu, ^{243}Am, ^{99}Tc, and ^{129}I combined. This is consistent with the dominance of the inhalation exposure pathway for the eruptive scenario and only a minor contribution from the groundwater pathway. At 10,000 years after closure (Figure 4.2-4), the igneous dose rate is dominated by groundwater releases, and the dose is dominated by ^{239}Pu, ^{237}Np, and ^{240}Pu. Additional contributions come from ^{99}Tc and ^{129}I. These results are consistent with the dominance of the groundwater pathway, and are similar to results for nominal performance at later times (see Section 4.1) because of the long half-lives of the major contributors. Contributions from ^{99}Tc and ^{129}I, which dominate the earliest releases in the nominal scenario (Figure 4.1.6), are relatively smaller in the igneous disruption scenario because the assumptions regarding drip shield and waste package performance (see Section 3.10.2) reduces the relative importance of the diffusive transport pathway.

4.2.3 Sensitivity of the TSPA-SR Results for the Igneous Disruption Scenario to Sample Size

Because of the approach taken to sampling on the time of intrusive events, the mean probability-weighted dose for igneous disruption is potentially sensitive to the number of realizations included in the analysis. For smaller sample sizes, individual realizations, as shown in Figure 4.2-1, have a potential to skew the mean upward or downward if, for example, combinations of parameter values that result in a high dose rate combine with early event times and high event probabilities or late event times and low event probabilities, respectively. Increasing the sample size provides a greater density of sampling on both event time and parameter stability, and provides a stable mean appropriate for comparison to the proposed regulatory limits.

Figure 4.2-5 shows a comparison of probability-weighted mean annual dose rates for igneous disruption calculated using three different sample sizes. The dose rate during the first 2000 years, when the curve is dominated by the eruptive releases, is essentially unchanged because of the very large number of eruptions included in the calculation (see Section 4.2.1.1). At later times, however, the curves differ due to the relative density of sampling on the time of the intrusive event. The blue curve, which is the case discussed in Section 4.2.2 and shown in Figure 4.2-1, uses 5,000 realizations to calculate performance for 50,000 years. This case has the highest density of sampling on the time of the event, and is the case used in Section 4.3 to construct the overall expected annual dose. The black curve in Figure 4.2-5 is calculated using 1,000 realizations for 100,000 years. The irregular shape of the curve in the first 20,000 years indicates that the sample size is too small, and that too few realizations are contributing to the determination of the mean during this interval. At later times, however, the probability that an intrusive event has already occurred increases, more realizations contribute to the mean, and the curve becomes smoother. The convergence of the 50,000 and 100,000 year cases between 30,000 and 40,000 years indicates that the sample size for the 100,000-year simulation is adequate at later times. The observation that the peak mean annual probability-weighted dose remains essentially constant after 50,000 years in the 100,000-year case provides a high degree of confidence that the overall expected annual dose rate, as described in Section 4.3, can be appropriately presented using results from the 50,000-year case.

The intermediate curve in Figure 4.2-5, shown in red, uses 1000 realizations to simulate the probability-weighted mean annual dose rate from igneous disruption during the first 20,000 years. This curve represents a smaller density of sampling on the time of the event than the 50,000-year case, but provides reasonable coverage during the period of primary interest. The near convergence of this curve with the two other curves as it approaches 20,000 years provides further confidence that it provides a reasonable estimate of performance. Because of the relative efficiency of this calculation and because of its emphasis on the period of greatest interest for igneous disruption, the 20,000-year case is used as a base case for comparison purposes in the sensitivity analyses reported in Section 5.2.9.

4.3 TOTAL SYSTEM PERFORMANCE FOR THE COMBINED SCENARIOS

As specified (see Section 1.3.3.2) in proposed 10 CFR 63.113(c) (64 FR 8640 [101680]), the TSPA for a compliance demonstration must consider all FEPs that have not been excluded through a systematic screening process (see Section 2.1). Specifically, the expected annual dose used for comparison with the dose limit in proposed 10 CFR 63.113(b) (64 FR 8640 [101680]) must be estimated "considering the probability of the occurrence of the events and the uncertainty, or variability, in parameter values used to describe the behavior of the geologic repository" (proposed 10 CFR 63.2 [64 FR 8640 [101680]). Thus, the expected annual dose includes all significant nominal and disruptive FEPs (i.e., both the likely behavior of the disposal system and the consequences of unlikely events), and the nominal and disruptive dose results must be combined to obtain the overall dose results specified in the proposed regulations.

In the TSPA&I issue resolution status report (NRC 2000 [149372], Section 4.4.1), the NRC provides guidance on the method of combining the expected annual doses for nominal and disruptive scenarios. The dose for each scenario is weighted by the scenario probability, so the summed expected annual dose includes both consequence and probability and therefore represents the expected risk for the repository (NRC 2000 [149372], Section 4.4.1). Consistent with the requirements of proposed 10 CFR 63.113(c) (64 FR 8640 [101680]), the expected annual dose histories do not include the human intrusion scenario. Results for the inadvertent human intrusion scenario are evaluated separately through a TSPA for a stylized human intrusion scenario, designed in accordance with NRC proposed requirements and guidance (Section 4.4).

4.3.1 Method for Combining Scenarios

As described in Sections 4.1 and 4.2, Monte Carlo simulations have been performed for the nominal and igneous-disruption scenarios. Within a scenario, uncertainty in input parameters and the future behavior of the system are addressed by computing many realizations, each of which has a different set of input parameters. As discussed in Sections 4.1.4 and 4.2.3, the base case includes 300 realizations of the nominal scenario and 5000 realizations of the igneous-disruption scenario. The large number of realizations needed for the igneous-disruption scenario is primarily driven by the need to have a fine-enough sampling on the time of the intrusive event (Section 4.2.3).

A straightforward method of combining the dose results for the nominal and disruptive scenarios (CRWMS M&O 2000 [147323], Section 4.4) is illustrated in Figure 4.3-1. In this method, first a conditional mean dose history is calculated for each scenario by averaging the dose values at each time step. Then, a weighted-mean dose history for each scenario is generated by multiplying the conditional mean dose by the scenario probability at each time step. Lastly, the final expected annual dose history is calculated by adding the two weighted-mean doses at each time step. This method is easily extended to any number of scenarios, but the illustration shows only two because those are the only scenarios that survived the scenario screening process (Section 2.1.1.1).

The method actually used for the TSPA-SR is slightly different (Figure 4.3-2). As discussed in Section 4.2, the igneous disruption simulations do not include nominal waste package failure processes; all releases are caused by igneous intrusion, directly or indirectly. Thus, the igneous disruption scenario differs slightly from the usual definition, in which all nominal processes are included in addition to the disruptive processes. It is important to note that for the regulatory period of 10,000 years there are no nominal waste package failures, so for that period, the igneous disruption scenario can be said to include all nominal processes. However, by 100,000 years many waste packages have failed by nominal processes (see Figure 4.1-9), and the igneous disruption results are significantly affected by the absence of nominal waste package failures.

Nominal processes are fully incorporated into the final expected annual dose by weighting the nominal scenario by 1 rather than by its actual probability, which is slightly less than 1. To explain further, let the probability of an igneous intrusion be p_i. Then the probability of no intrusion, which is to say the probability of only nominal processes, is $1 - p_i$. At a particular time, denote the nominal annual dose by D_n and the disruptive annual dose (not including nominal failures) by D_i. Because it is assumed that nominal models can be used in simulating the igneous disruption scenario (Section 3.10), the annual dose for an igneous disruption, including all nominal processes, is approximated by $D_n + D_i$. This approximation is an equality if there is no overlap between the waste packages that fail because of nominal processes and the waste packages that fail because of the igneous event. The approximation is conservative if there is an overlap because, in that case, releases from one or more waste packages are being counted twice. The probability-weighted total annual dose is then approximated by $D_t = (1 - p_i)D_n + p_i(D_n + D_i) = D_n + p_iD_i$. This equality shows that if the nominal scenario is weighted by 1 and the igneous disruption scenario without nominal failures is weighted by the disruption probability, then the correct expected annual dose will be calculated (Figure 4.3-2).

Another difference between the method illustrated in Figure 4.3-1 and the method used for this TSPA (shown in Figure 4.3-2) is that the conditional mean dose history for the igneous disruption scenario is never calculated. The probability weighting is applied individually to each realization, and each realization has a different probability weight, as discussed in Section 3.10. This variant of the method is applied because the probability of an igneous intrusion is considered to be uncertain and is sampled from a probability distribution. The net effect is still to weight each dose history by its appropriate probability to obtain the proper expected annual dose.

4.3.2 Results for Combined Performance

Results for the nominal scenario are discussed in Section 4.1, with the dose histories and some statistical measures (mean, median, 5th percentile, and 95th percentile) given in Figure 4.1-5. Results for the igneous disruption scenario (with nominal waste package failures excluded) are discussed in Section 4.2, with the dose histories and statistical measures given in Figure 4.2-1. For both the nominal and disruptive scenarios, the results given previously already have the proper weighting. The nominal doses shown in Figure 4.1-5 are conditional, so they have the full probabilistic weighting of 1. The igneous doses shown in Figure 4.2-1 are already weighted by the probability of an igneous event. Thus, the combined dose for the total system is obtained simply by adding them. Because the nominal and disruptive Monte Carlo simulations do not use the same values for parameters that are common to both (because of their different numbers of realizations), the individual realizations can not be added. However, the correct mean annual dose is obtained by adding together the two scenario mean curves, as discussed above. The combined mean dose curve, along with the mean curves for the two individual scenarios for comparison, are shown in Figure 4.3-3. The combined mean curve in Figure 4.3-3 corresponds to the expected annual dose, which is the performance measure of proposed 10 CFR 63.113(b) (64 FR 8640 [101680]).

At early times (before about 40,000 years), igneous disruption doses dominate the scenario combination. In particular, the combined dose is given by igneous disruption alone during the 10,000-year regulatory period, because there are no nominal waste package failures that early.

However, during later times (after about 40,000 years), the combined dose is dominated by the nominal dose because of the large number of nominal waste package failures. The base-case igneous disruption scenario was simulated for only 50,000 years. However, by 50,000 years the sum is dominated by the nominal scenario, so the combined mean curve is continued out to 100,000 years by neglecting the contribution from igneous intrusion after 50,000 years.

The detail regarding radionuclide contributions to dose and intermediate subsystem results contained in Sections 4.1 and 4.2 will not be duplicated here. The radionuclide and subsystem results described in Section 4.2 apply as well to the combined system at early times, and the radionuclide and subsystem results described in Section 4.1 apply as well to the combined system at later times.

4.4 INTRUSION HUMAN PERFORMANCE RESULTS

Proposed 10 CFR Part 963 (64 FR 67054 [124754], Sections VI.B(g), VI.B(h), 963.16, and 963.17) includes a requirement for the DOE to conduct a separate performance assessment to evaluate the consequences of human intrusion (see Section 1.3.1.4). The separate human intrusion performance assessment must be evaluated in terms of compliance with the same radiation protection standard considered for the combined nominal and disruptive performance assessment described in Section 4.3 (Figure 4.4-1).

Proposed 10 CFR Part 963 (64 FR 67054 [124754], Section 963.16[a][2]) states that the human intrusion evaluation should be "consistent with applicable NRC regulations regarding a stylized human intrusion case." The applicable NRC regulations are contained in proposed 10 CFR Part 63 (64 FR 8640 [101680], Sections XI, 63.102, and 63.113). Proposed 10 CFR Part 963 (64 FR 67054 [124754], Section II.K.1) also states that "it is anticipated that NRC would conform its proposed licensing regulation at 10 CFR Part 63 (64 FR 8640 [101680]) to the final EPA radiation protection standards, as necessary and appropriate." The EPA radiation protection standards are contained in proposed 40 CFR Part 197 (64 FR 46976 [105065], Sections III.C, III.E, 197.25, and 197.26). As discussed in Section 1.3 (see Table 1.3-3), the proposed NRC regulation and the proposed EPA standard are not identical for human intrusion, although they must be consistent when finally promulgated. Therefore, it is necessary to consider aspects of human intrusion contained in both the NRC regulation and the EPA standard that might logically be incorporated when they are finally promulgated.

The details of the stylized human intrusion scenario, based on the NRC regulations, are described in Section 4.4.1. Aspects of human intrusion contained in the proposed EPA radiation protection standards that differ from the proposed NRC regulations are also discussed. The results of the human intrusion performance assessment and comparison with the radiation protection standards are presented in Section 4.4.2.

4.4.1 Technical Bases for Human Intrusion Analyses

4.4.1.1 Regulatory Basis for Human Intrusion

The flow of information wheel for the stylized human intrusion scenario is shown in (Figure 4.4-2). It is based on the current TSPA-SR conceptualization of the human intrusion assumptions that are outlined in both the proposed NRC regulation and EPA standard. These assumptions, and resolution of any differences, are summarized in Table 4.4-1 (also see Table 1.3-3).

Table 4.4-1. Human Intrusion Scenario Regulatory Assumptions

NRC Base Assumptions (from Proposed 10 CFR Part 63 [64 FR 8640 [101680])	EPA Additional and/or Conflicting Assumptions (from Proposed 40 CFR Part 197 [64 FR 46976 [105065])	Conceptualization for TSPA-SR
Assumed intrusion is a drilling event.	Assumed intrusion is acute and inadvertent.	Inadvertent drilling event.
Drilling result is a single, nearly vertical borehole that penetrates a waste package and extends down to the SZ.	Borehole penetrates a degraded waste package.	Single vertical borehole from surface through a single waste package to the SZ.
Intrusion occurs 100 years after closure.	Intrusion time should take into account the earliest time after disposal that a waste package could degrade sufficiently that current drilling techniques could lead to waste package penetration without recognition by the drillers.	Intrusion occurs at 100 years (a 10,000 year intrusion time is examined in a sensitivity simulation).
Borehole properties (diameter, drilling fluids) are based on current practices for resource exploration.	Borehole results from exploratory drilling for groundwater.	Borehole diameter consistent with an exploration groundwater well.
Borehole is not adequately sealed to prevent infiltrating water.	Natural degradation processes gradually modify the borehole, the result is no more severe than the creation of a groundwater flow path from the crest of Yucca Mountain through the potential repository and to the water table.	Infiltration and transport through the borehole assumes a degraded, uncased borehole, with properties similar to a fault pathway.
Hazards to the drillers or to the public from material brought to the surface by the assumed intrusion should not be considered.	Only consider releases through the borehole to the SZ; consider releases occur gradually through air and water pathways, not suddenly as with direct removal.	Groundwater is only pathway considered.
A separate consequence analysis is required, identical to the performance assessment, except for the occurrence of the specified human intrusion scenario.	Unlikely natural processes and events are not included, but analysis could include disturbances by other processes or events that are likely to occur.	Intrusion borehole is applied to nominal case; effects of volcanism are not included.
Peak dose is not to exceed 25 mrem/yr. in the first 10,000 years.	Peak dose is not to exceed 15 mrem/yr. in the first 10,000 years.	Does not affect simulations.

A key difference noted in Table 4.4-1 between the NRC and EPA regulations is the time of intrusion. In proposed 10 CFR 63.113 (d) (64 FR 8640 [101680]), the NRC assumes the intrusion to occur 100 years after permanent closure (Figure 4.4-3 (a)). The EPA states in proposed 40 CFR 197.26 (g) (64 FR 46976 [105065]) that the intrusion occurs at a time determined by the NRC, but also states in proposed 40 CFR 197.25 and 197.26 (g) (64 FR 46976 [105065]) that the NRC should take into account the earliest time after disposal that a waste package could degrade sufficiently that current drilling techniques could lead to waste package penetration without recognition by the drillers (Figure 4.4-3 (b)). The EPA (64 FR 46976 [105065], Section III.E) further points out that a waste package would likely be recognizable to a driller for "at least thousands of years" and that if the intrusion could not occur until after

10,000 years that the results would not be part of the licensing process but would be included in the Yucca Mountain EIS instead.

For TSPA-SR, human intrusion was assumed to occur at 100 years after closure into a relatively intact waste package. This early intrusion time was selected because it was considered to be conservative and because it was difficult to defensibly quantify a later intrusion time consistent with EPA guidance. A later intrusion time (10,000 years), approximating the EPA guidance, was examined in a sensitivity analysis (see Section 4.4.2).

The construction of the human intrusion conceptual model for TSPA-SR, based on the regulatory guidance summarized in this section, is described in Section 4.4.1.2. The implementation and integration of the human intrusion model in TSPA-SR is described in Section 4.4.1.3.

4.4.1.2 Features, Events, Processes, and Conceptual Model for Human Intrusion

A comprehensive set of FEPs relevant to the human intrusion scenario were evaluated (see Appendix B). The FEPs directly relevant to the human intrusion event are listed in Table B-9 (all FEPs with a FEP number beginning with 1.4). In general, the human intrusion FEPs were screened in or out of the TSPA-SR based on the regulatory guidance summarized in Table 4.4-1. For excluded (screened out) FEPs, the basis for exclusion is summarized in Appendix B. Those FEPs that are included (screened in) in the human intrusion conceptual model (as listed in Table B-9) along with the regulatory guidance in Table 4.4-1, form the basis for the TSPA-SR human intrusion conceptual model (Figure 4.4-4).

The TSPA-SR human intrusion scenario conceptual model includes five key components (Figure 4.4-5):

- Infiltration of water down the borehole and into the penetrated waste package
- Mobilization and release of radionuclides from the penetrated waste package
- Transport of radionuclides down the borehole to the water table
- Transport of radionuclides through the SZ
- Biosphere exposure pathways and dose calculation at the receptor location.

The implementation and integration of these five key components of human intrusion into the TSPA-SR model are described in Section 4.4.1.3.

4.4.1.3 Implementation and Integration of Human Intrusion within Total System Performance Assessment

As identified in Table 4.4-1, the human intrusion scenario is identical to the nominal scenario, except for the intrusion borehole. Therefore, the TSPA-SR model for human intrusion derives from the TSPA-SR nominal scenario model. The human intrusion borehole is assumed to be drilled from the ground surface (at a random location within the footprint of the potential repository), through the drip shield and a single waste package (top and bottom), to the water table. The borehole is assumed to have the diameter of a standard rock bit used for water well drilling.

Three of the key human intrusion components identified in Section 4.4.1.2—infiltration down the borehole, waste mobilization and release, and borehole transport—are associated with the intrusion borehole. In the TSPA-SR human intrusion model, infiltration down the borehole and borehole transport replace the performance contribution provided by the UZ in the nominal scenario. Implementation of waste mobilization and release in the human intrusion model differs slightly from the nominal model because the failure mode (drilling breach rather than corrosion) and number of packages failed (one rather than many) are different. All of the waste within the penetrated waste package is assumed to have cladding perforations from the action of the drillbit.

Implementation of the other two key human intrusion components—SZ transport and biosphere—is consistent with the nominal model. Wherever possible, nominal model components and input data for climate, SZ transport, and biosphere processes were used.

The TSPA-SR human intrusion model was used to simulate a 100,000-year period using climatic and hydrogeologic conditions consistent with the nominal scenario. Consistent with the regulatory definition of a TSPA (64 FR 67054 [124754], Section 963.2), the human intrusion model was run probabilistically.

The five key components of human intrusion described in Section 4.4.1.2 are based on FEPs screening and regulatory guidance. However, to fully define the human intrusion scenario for implementation within TSPA-SR, a number of additional technical assumptions, not addressed in the regulations, were required. The technical assumptions are summarized in Table 4.4-2. Key technical assumptions are shown schematically in Figure 4.4-6.

Table 4.4-2. Human Intrusion Scenario Technical Assumptions

Issue	Key Component Affected	TSPA-SR Implementation
Borehole diameter	Infiltration Borehole Transport	Typical water well borehole has a diameter of 20.3 cm (8 in.).
Infiltration into borehole	Infiltration	Assumed infiltration rate distribution is based on modeled infiltration in the Yucca Mountain region for the glacial transition climate. Values at the high end of the distribution inherently include the possibility of surface water collection basin focusing.
Seepage into penetrated waste package	Infiltration Waste Mobilization	Volumetric flux is equivalent to infiltration rate times borehole area. Volume of drilling fluid is ignored.
Type of waste package penetrated	Waste Mobilization	Sampled from CSNF and co-disposed waste packages. Co-disposed packages contain both DSNF and HLW glass.
Thermal and geochemical conditions in waste package	Waste Mobilization	Assume temperature and in-package chemistry as calculated in nominal scenario. This assumes Well J-13 water and ignores any chemical effects of the drilling fluid.
Waste form degradation	Waste Mobilization	Waste in penetrated package is assumed to have perforated cladding from drilling disturbance.
Solubilization of radionuclides in water	Waste Mobilization	Infiltrating water can mix with waste in entire waste package. Solubility is based on temperature and in-package chemistry as in nominal scenario.

Table 4.4-2. Human Intrusion Scenario Technical Assumptions (Continued)

Issue	Key Component Affected	TSPA-SR Implementation
Borehole flow and transport properties	Infiltration Borehole Transport	Volumetric flux consistent with seepage into the waste package. Transport properties consistent with a UZ fault pathway.
Borehole location	Infiltration SZ Transport	Random over the footprint of the potential repository. Uncertainty in location is captured in infiltration rate and location that radionuclides enter the SZ.
Borehole length	Borehole Transport	Borehole length from the potential repository to SZ conservatively assumes water level consistent with glacial transition climate.
SZ	SZ Transport	Assume SZ flow and transport properties identical to nominal scenario.
Biosphere processes	Biosphere	Assume exposure pathways and receptor characteristics identical to nominal scenario.

The implementation and integration of the five key components and the associated technical assumptions in the TSPA-SR human intrusion model are described in more detail in the following five subsections. The model implementation is shown schematically in Figure 4.4-7. A more detailed description of specific input parameters and their values is presented in the TSPA-SR Model (CRWMS M&O 2000 [148384], Section 6.3.9.3).

4.4.1.3.1 Infiltration Down Borehole

The inputs, outputs, and technical assumptions associated with the infiltration of water down the borehole and into the penetrated waste package are summarized in Figure 4.4-8. The volume of water flowing down the intrusion borehole and into the waste package was based on infiltration rather than precipitation. It was deemed unrealistic to assume that all precipitation would flow directly down a degraded borehole. Instead, infiltrating water was assumed to flow preferentially down the degraded borehole, bypassing the UZ. The infiltration rate was sampled from a distribution based on the modeled infiltration calculated for the glacial transition climate (CRWMS M&O 2000 [145774], Section 3.5.2). Infiltration rates at the upper end of the distribution account for the possibility of enhanced infiltration if the borehole is located in an area where it might capture significant runoff (e.g., a wash or other surface water collection basin). The volumetric flux of water down the borehole is based on the infiltration rate entering a 20.3-cm-diameter borehole. The calculated volumetric flux down the borehole from surface infiltration was assumed to flow directly into the penetrated waste package, with no gain or loss from the surrounding UZ.

In the TSPA-SR, the infiltration down the borehole was not explicitly modeled, rather the borehole infiltration flux was assumed to flow directly into the penetrated waste package. The effect of a higher infiltration flux (with the flux fixed at the 95th percentile value) was examined in a sensitivity simulation (see Section 5.2.10).

4.4.1.3.2 Waste Mobilization and Release

The inputs, outputs, and technical assumptions associated with the mobilization and release of radionuclides from the penetrated waste package are summarized in Figure 4.4-9. The volumetric flux of water from infiltration (described in Section 4.4.1.3.1) was assumed to flow directly into the penetrated waste package, with no resistance from the drip shield or waste package. The penetrated waste package was represented by a mixing cell in which the entire volume of waste from the waste package was available for degradation. The waste package type was sampled based on the relative number of CSNF and co-disposed waste packages present in the inventory. Under the current thermal loading, the penetrated waste package would be thermally hot at 100 years to the extent that liquid water entering the waste package would be converted to vapor. Therefore, a conservative bias is incorporated into the human intrusion scenario by assuming that liquid water flows through the penetrated waste package and dissolves radionuclides at 100 years.

Waste mobilization and release processes are not specified in regulations. It was assumed that all waste in the penetrated package had cladding perforated by drillbit damage, enhancing mobilization. However, no waste was assumed to be transported directly to the SZ (e.g., as cuttings on the drillbit or by falling to the bottom of the borehole before it degrades) or to the surface. The radionuclides in the waste were assumed to be mobilized by advection (mixing with and dissolution in the infiltrating water) and diffusion. Radionuclide solubilities were based on nominal scenario in-package temperature and chemistry. This assumes Well J-13 water, which inherently ignores the chemical effects of the drilling fluid.

As described in Section 3.5.1.1, the human intrusion analysis used the radionuclides from the nominal scenario along with two additional radionuclides: ^{90}Sr and ^{137}Cs. The additional radionuclides were screened out of the nominal scenario based on short half-lives and/or high sorption in the UZ. With a human intrusion at 100 years that bypasses the UZ, they were potentially important to the human intrusion scenario. The human intrusion analyses include transport of both dissolved species and colloidal particles. This suite of radionuclides adequately encompasses doses that could result from both dissolved and colloidal transport of radionuclides.

All of the mobilized radionuclides were assumed to be released through the bottom breach in the waste package for advective transport down the borehole to the SZ.

4.4.1.3.3 Borehole Transport

The inputs, outputs, and technical assumptions associated with the transport of radionuclides down the borehole to the water table (i.e., to the SZ) are summarized in Figure 4.4-10. The radionuclide mass (both dissolved and colloidal) released from the waste package (described in Section 4.4.1.3.2) was assumed to be transported down the human intrusion borehole to the water table. At the water table, the radionuclide mass was transported by processes appropriate for the SZ (see Section 4.4.1.3.4). As described in Table 4.4-1, the borehole transport pathway was assumed to be a degraded, uncased borehole, with properties similar to a fault pathway in the UZ.

The implementation of the borehole transport pathway below the potential repository in TSPA-SR is a one-dimensional "pipe" pathway. The pipe pathway requires borehole cross-sectional area, volumetric flux, porosity, fluid saturation, and dispersivity. The volumetric flux was assumed to be the same as in the borehole above the waste package. The pipe transport included sorption (based on devitrified unit properties), but did not include matrix diffusion.

The borehole length from the potential repository to the SZ of 190 m (623 feet) conservatively assumed a water level consistent with the glacial transition climate. The distance from the potential repository to the water table is greater for the present day and monsoon climates.

4.4.1.3.4 Saturated Zone Transport

The transport of radionuclides through the SZ in the TSPA-SR human intrusion model was identical to the nominal model, with two exceptions. First, two additional radionuclides (^{90}Sr and ^{137}Cs) were added. Second, the radionuclide source region for the SZ was sampled between regions 1 and 3; regions 2 and 4 were never used (CRWMS M&O 2000 [139440], Figure 2).

Because the intrusion borehole could be located anywhere within the footprint of the potential repository, the radionuclides transported down the borehole could enter the SZ at any point below the footprint with equal probability. Sampling between regions 1 and 3 (the two SZ source regions that underlie the footprint) accounts for the randomness in the location of the intrusion borehole.

4.4.1.3.5 Biosphere Processes

The biosphere exposure pathways and dose calculation at the receptor location in the TSPA-SR human intrusion model were identical to the nominal model, except that additional BDCFs were required to account for the two additional radionuclides.

4.4.2 Results and Interpretation of Human Intrusion Analyses

The regulations require a separate performance assessment to evaluate the consequences of human intrusion. The separate human intrusion performance assessment was evaluated in terms of compliance with the same radiation protection standard considered for the combined nominal and disruptive performance assessment.

As with the combined nominal and disruptive scenario, the human intrusion scenario was run probabilistically. Uncertainty is explicitly included in the models and input parameters in the form of discrete probability distributions. Uncertainty in the possible performance was evaluated through the use of these probabilistic analyses. Although the expected (or mean) human intrusion performance of potential repository system can be determined from the range of probabilistic outcomes, the entire range of possible outcomes was examined.

Figure 4.4-11 shows the TSPA results for the base case human intrusion scenario with an intrusion at 100 years after closure. As with the combined nominal and disruptive performance assessment, the time period of regulatory interest is 10,000 years after the closure of the potential repository. This figure illustrates the human intrusion dose that is projected to occur out to

100,000 years to provide a level of confidence that there are no unexpected changes in dose beyond the regulatory period of performance.

Figure 4.4-11 shows 300 simulated dose histories along with some statistical measures of the dose distribution. The mean curve is generated by averaging the 300 dose values at each time step, and the percentile curves are generated by determining the location of the given percentile at each time step (for example, the median curve is generated by determining the dose which has half of the calculated doses above it and half below it at each time step). There is a considerable amount of variability in the projections of human intrusion dose, and there are non-zero doses within the first 10,000 years. However, no dose for any of the 300 realizations exceeds 0.5 mrem/yr over the first 10,000 years. The peak mean human intrusion dose during the first 10,000 years after potential repository closure is approximately 0.008 mrem/yr., occurring at approximately 1,000 years. Over the entire 100,000 years, the peak mean dose is also approximately 0.008 mrem/yr. and the peak median dose (50 percent probability) is about 0.00007 mrem/yr. The mean dose rate is significantly larger than the median dose rate because the mean is dominated by a few realizations with high dose rate histories, while the median represents the mid-point results from all 300 realizations.

Figure 4.4-12 shows the TSPA results for the human intrusion scenario with an intrusion at 10,000 years after closure. This simulation examines the effect of an intrusion at a hypothetical time more consistent with the EPA regulation (see Table 4.4-1). Figure 4.4-12 shows a comparison of the mean human intrusion dose curve from an intrusion at 10,000 years with the mean dose curve from the base case intrusion at 100 years. For the intrusion at 10,000 years, there were no doses prior to 10,000 years and the peak mean dose over 100,000 years is less than for the base case intrusion at 100 years. At 100,000 years, the mean dose is nearly identical to the mean dose from the base case.

The analyses compared in Figure 4.4-12 were performed using probabilistic TSPA simulations with 100 realizations. One hundred realizations were used rather than the 300 used for the base-case simulation because 100 realizations are sufficient to see the relative effects when comparison is made to the base case.

4.5 TREATMENT OF POTENTIALLY DISRUPTIVE EVENTS IN THE TOTAL SYSTEM PERFORMANCE ASSESSMENT

The following sections describe the TSPA-SR treatment of specific potentially disruptive events identified in the Repository Safety Strategy Rev. 3 (CRWMS M&O 2000 [139593], Section 2.3) and Rev. 4 (CRWMS M&O 2000 [148713], Section 5.3) as being of special concern. These potentially disruptive events fall into two categories:

- Those leading to extreme environments or conditions that could affect any one of the principal factors that determine postclosure performance

- Those directly disrupting important potential repository system barriers, including

  - Inadvertent human intrusion
  - Rise of the water table

- Seismic activity
- Igneous activity
- Waste-generated disruptions (radiolysis, pyrophoricity, nuclear criticality)
- Early failure of engineered barriers
- Drift collapse.

As described in Section 2.1.1.1, the TSPA-SR has defined disruptive events (as distinguished from potentially disruptive events) to be those events that have a probability less than one and that have not been excluded by the FEPs screening process. This definition was chosen to be consistent with the FEP screening process, and is somewhat inconsistent with other usages that define events as disruptive prior to the evaluation of their impact on overall performance. For example, the *Disruptive Events Process Model Report* (CRWMS M&O 2000 [141733]) addresses only those FEPs related to igneous activity, tectonic activity, and seismicity, and includes consideration of FEPs that are screened out of the TSPA on the basis of low probability (e.g., faulting) or low consequence (e.g., long-term tectonic changes), as well as FEPs that are included in the nominal scenario class (seismically-induced cladding failure) and the disruptive scenario class (igneous activity). Proposed 10 CFR Part 963.117 (64 FR 67054 [124754]) specifies consideration of four disruptive events: volcanism, seismic events, nuclear criticality, and inadvertent human intrusion. Each of these four events is treated differently in the TSPA-SR: volcanism, seismicity, and criticality have been examined through the FEPs process, and only volcanism has been retained for analysis in a separate disruptive scenario. Human intrusion has been treated through a separate performance assessment analysis (Section 4.4), consistent with proposed regulatory requirements. Other potentially disruptive events have been evaluated in detail through the FEPs screening process (see Section 2.1 and Appendix B) and have been excluded from the TSPA-SR on the basis of the probability or consequence criteria defined by the applicable proposed regulations. The YMP FEP database (CRWMS M&O 2000 [150806], Appendix D) provides detailed documentation of the screening of all potentially disruptive events, including many not listed above. The treatment of the potentially disruptive events listed above is discussed in the following sections.

4.5.1 Extreme Conditions Affecting Principal Factors

As described in the Repository Safety Strategy Rev. 3 (CRWMS M&O 2000 [139593], Section 2.3.1), extreme conditions affecting any one of the principal factors may include, for example, potentially high local flow conditions or extreme water chemistries that could affect waste package performance. In general, extreme environmental conditions that are within the range of reasonable and realistic possibilities for the future have been included in the models and parameter distributions used to characterize the nominal performance of the potential repository, as described in Section 3. The probabilistic approach used in the TSPA-SR uncertainty analysis (Section 2.2.4) ensures that effects of these extreme conditions are included in the analysis. Sensitivity analysis techniques (Section 2.2.5) allow recognition of the impact of extreme parameter values on overall performance. Results of the TSPA-SR uncertainty analysis are described in Section 5.1. In some cases, additional sensitivity analyses have been performed using specified parameter values, rather than distributions, to provide insight into the importance of parameters representing processes of interest. These sensitivity analyses are described in Section 5.2.

4.5.2 Barrier Disruption Due to Inadvertent Human Intrusion

Because future human activity cannot be predicted, the possibility of inadvertent intrusion into the potential repository in the future cannot be precluded. The NRC and the EPA have each recognized this possibility, and they have proposed requiring that the DOE assess potential repository performance following a stylized drilling intrusion, such as might occur during exploration for water or other natural resources. An analysis of the consequences of human intrusion, as prescribed in the proposed NRC and EPA regulations, is presented in Section 4.4. In addition, studies have been conducted regarding the potential for future exploration at the site. Extensive review of the potential for occurrences of natural resources at Yucca Mountain has concluded that no currently economic resources occur at the site, nor are any likely to be found in the future (DOE 1998 [100548], Section 2.2.7.3). Regardless of the specification to consider human intrusion by drilling, the probability of future drilling for natural resources at Yucca Mountain is extremely low.

4.5.3 Barrier Disruption Due to Water Table Rise

Various mechanisms have been proposed that could cause changes in the elevation of the water table, including climate change, seismic pumping, changes in regional stress, and hydrothermal effects associated with igneous activity. Water table changes due to climate change have been included in the nominal scenario through changes in the SZ groundwater flow model (Section 3.8). Other causes of water table rises have been evaluated through the FEPs screening process and have been shown to be of sufficiently low consequence or probability to have been excluded from the TSPA analysis (see Appendix B and CRWMS M&O 2000 [150806], Appendix D).

4.5.4 Barrier Disruption Due to Seismic Activity

Seismic events of uncertain magnitude are likely to occur at Yucca Mountain in the future, and seismicity is therefore more appropriately thought of as an expected, or nominal, process rather than as a disruptive event. Various consequences of seismic activity and faulting have been considered in detail for the TSPA-SR, including direct damage to drifts due to fault displacement; effects of fault displacement on groundwater flow; damage to drip shields, waste packages, and the waste itself from vibratory ground motion; and damage to drifts and engineered barriers from rock falls induced by ground motion. Damage to fuel rod cladding due to seismic ground motion is explicitly included in the TSPA-SR, as described in Section 3.5.4. All other consequences of ground motion have been excluded from the TSPA-SR on the basis of low consequence or low probability (see Appendix B and CRWMS M&O 2000 [150806], Appendix D).

4.5.5 Barrier Disruption Due to Igneous Activity

Igneous activity has been included in the TSPA-SR explicitly in the disruptive scenario class. Section 3.10 describes the TSPA models for the probability and consequence of igneous activity at the site, and Section 4.2 describes the results of the TSPA-SR analysis of igneous activity. As described in Section 3.10.2, not all FEPs potentially relevant to igneous activity have been included in the model. Instead, the model focuses on analyzing two igneous disruption scenarios: (1) the direct release of waste during an eruption and the subsequent atmospheric transport in an ash plume and (2) the release and subsequent groundwater transport of radionuclides from packages that are damaged by intrusion. Other potentially relevant events and processes, including those associated with the effects of intrusions that do not intersect the potential repository, have been evaluated and excluded from the TSPA on the basis of low consequence or probability (see Appendix B and CRWMS M&O 2000 [150806], Appendix D).

4.5.6 Barrier Disruption Due to Waste-Generated Changes

Waste-generated changes include FEPs related to repository-generated thermal effects on fluid flow and chemistry and FEPs related to waste-specific processes such as radiolysis, pyrophoricity, and nuclear criticality. Most waste-generated changes are included in the uncertainty analysis of nominal performance through models and parameters that characterize thermal, hydrologic, and chemical effects. Radiolysis and pyrophoricity have been evaluated and have been excluded from the TSPA-SR on the basis of low consequence or low probability (see Appendix B and CRWMS M&O 2000 [150806], Appendix D).

Nuclear criticality was evaluated in *Disposal Criticality Analysis Methodology Topical Report* (YMP 1998 [104441]). Criticality in the waste and the engineered barrier system (EBS) has been excluded from the TSPA-SR on the basis of low probability of occurrence during the first 10,000 years of performance. As described in *Probability of Criticality Before 10,000 Years* (CRWMS M&O 2000 [149939]), the probability of a nuclear criticality event at Yucca Mountain has been examined under conditions of nominal performance, potential damage due to seismicity, and igneous disruption. The probability of criticality in the waste package, near-field, and far-field has been shown to be below one chance in 10,000 in the first 10,000 years following repository closure for nominal performance for all waste types. This conclusion is based on the low probability of waste package failure during the first 10,000 years (waste package failure is a necessary condition for all configurations that could lead to criticality), and includes consideration of potential seismic effects. For the igneous disruption scenario, the probability of criticality in commercial spent nuclear fuel (CSNF) both within partially damaged packages and in fuel/magma mixtures that might occur following complete damage of packages are also shown to be below one chance in 10,000 in 10,000 years. The screening decision is preliminary because calculations are incomplete for criticality of DSNF in magma following igneous disruption, and for criticality events of all waste types outside the package following igneous disruption, in both the near-field and far-field.

4.5.7 Early Failure of Engineered Barriers

Waste packages and drip shields have a potential to fail earlier than might be predicted simply as a result of corrosion processes for a variety of reasons, including manufacturing defects, damage during shipment, improper placement, or other factors that escape detection during the operational period. As discussed in Section 3.4, manufacturing defects in the waste packages are included in the modeling of waste package performance for the nominal scenario, and are shown to have little effect on failure due to corrosion. Manufacturing defects in the drip shield are excluded from the TSPA-SR due to the low probability that significant defects will remain following construction of the drip shield components. Defects associated with errors in waste package and drip shield emplacement (and other operational activities) have been excluded from the TSPA on the basis of low probability that they will escape detection during operational phase inspections (CRWMS M&O 2000 [147359]).

4.5.8 Barrier Disruption Due to Drift Collapse

Degradation of the ground support system and subsequent rockfall are probably more appropriately thought of as part of the nominal evolution of the potential repository (and any underground mined facility), rather than as disruptive events. The timing and magnitude of rockfall are uncertain, but the eventual occurrence of some rockfall and drift degradation is highly likely. Analyses of the joint orientation and spacing in the underground support system define the range of block sizes expected; analyses of the response of waste packages and drip shields to this range of rockfalls indicates that rockfalls will not significantly affect performance of the engineered barriers, with or without backfill. Rockfall and drift collapse have, therefore, been screened out of the TSPA-SR analysis on the basis of low consequence (see Appendix B and CRWMS M&O 2000 [150806], Appendix D).

4.6 ALTERNATIVE REPOSITORY DESIGN ANALYSES

The SR reference repository design is presented in Section 1.7. One of the aspects of the reference design is the concept that no backfill will be emplaced over the drip shields. The analyses presented in this section include comparisons between the reference design and the two following alternative design cases: (i) the reference design with backfill (partially filling the drift over the drip shield) and (ii) a low thermal load design. These comparisons evaluate the effect of each alternative design case on waste package temperature profiles, waste package failure, and overall performance of the potential repository. Note that the backfill design alternative causes an increase in internal waste package temperatures above design goals, making it unacceptable as a reference design.

4.6.1 Reference Design with Backfill

The backfill alternative (CRWMS M&O 2000 [151014]) is intended to provide some protection of the drip shield from rockfall and to reduce the amount of liquid water in contact with the drip shield and waste package, which is an important factor impacting waste package lifetimes. Backfill may cause a reduction in relative humidity for longer periods after waste emplacement, thus causing further delay in the initiation of corrosion. Backfill also may be considered to control the chemistry of the system, thus controlling waste package corrosion, waste form dissolution, EBS transport, and to some extent geologic transport.

The design for covering the drip shields with backfill is shown in Figure 4.6-1. This design was incorporated into the thermal hydrologic model, the drip shield degradation model, and the waste package degradation model to determine the effect on waste package and, ultimately, repository system performance. The backfill was assumed to have the thermal properties of crushed tuff, to be dry at emplacement, and to be introduced 50 years after waste emplacement. The initial moisture in the backfill may cause early corrosion, but this effect is expected to be small as temperatures in the potential repository quickly rise to drive off the moisture in the backfill. The primary effect on the system is an increase in temperature at the time of backfilling and a corresponding delay in the relative humidity increase as the potential repository cools. While the effect of moisture control is an important potential benefit of backfill, no change in seepage contacting the waste package as a result of backfilling was included in the modeling. The full evaluation of the effects of backfill on system performance have not been completed because the effects of backfill on seepage have not been quantified and included in the analysis. However, it is expected that the seepage implemented in this model is conservative because the seepage is all expected to contact the waste package and not be redirected around the waste package due to backfill. For the model assumptions included herein, there is little difference in total system performance between backfill and no-backfill cases.

The thermal hydrologic results comparing the case with the reference design (no backfill in the emplacement drift) and a backfilled emplacement drift are illustrated in Figure 4.6-2. The large temperature increase at 50 years corresponds to the emplacement time of the backfill. Dramatic increases of temperature in waste package, drip shield, and invert are observed from the no-backfill to the backfill cases. Average peak temperature of the waste package increases from 160 to 280°C. The higher temperature of the backfill case is maintained up to 2,000 years. The top of the drip shield demonstrates a pattern of temperature increase similar to the waste package. The sharp increase of temperature was expected because the backfill should localize heat near the waste package. For an emplacement drift without backfill, the dominant mode of heat transfer is radiation, which is an efficient mode of heat transfer between the waste package surface and the drift wall. For a backfilled drift the dominant mode of heat transfer is no longer thermal radiation, the resistance to heat transfer from the waste package surface to the drift wall is greatly increased; therefore, the waste package temperature increases. Accompanying the increase in surface temperature is a corresponding decrease in relative humidity in the drift. These results are presented in Figure 4.6-3.

The initial waste package failure curves are shown in Figure 4.6-4 for the no-backfill case and the backfill case. The curves show that there is not a significant difference in the two cases as they are currently modeled. This leads to the conclusion that the overall dose comparison between the two cases will not show a significant difference either.

The effects of backfill in the drift wall are minimal in the thermal hydrologic sensitivity analysis. The temperature only increases 16°C of averaged peak temperature, and a small amount of relative humidity decrease is noted regarding with the increase of temperature. However, because the backfill decreases the heat transfer to the drift wall, which creates large thermal gradients within the drift, the predicted drift wall temperatures tend to be hotter due to scaling difficulties between the mountain scale and drift scale models. Even considering the scale effects, the effects of backfill for the drift wall are still insignificant for the performance of the potential repository (see Figure 4.6-5).

In conclusion, the backfill design localizes the heat near the waste package and decreases relative humidity for a considerable time which could delay waste package degradation. However, the effects of backfill are diminished even at the drift wall because the temperature and the relative humidity changes at that location are insignificant.

The effect of backfill on seepage is uncertain, and additional testing is being conducted to better understand these effects. As noted, the backfill is assumed in these analyses not to alter the seepage onto the waste package. Some of the potential effects of backfill on seepage are as follows:

- Diversion of the seepage around the waste package
- Reduction of seepage reaching waste package because of evaporation of incoming water that is at a low flow rate
- Concentration of salts in the backfill from seepage water
- Promotion of water condensation at the contacts between the backfill and the waste package surface
- The backfill is not expected to restore the preconstruction UZ flow conditions. These effects have not been fully evaluated for their impact on overall system performance.

The backfill potentially has an impact on the conditions in the disruptive event or volcanism scenario. The backfill may tend to clog or slow the intrusion of the magma into the drift, thus preventing damage to as many waste packages as may be damaged if the magma flows unimpeded down a drift. The difference in the dose as modeled in the disruptive scenario is shown in Figure 4.6-6. The relatively small difference in performance between the backfill case and the no-backfill case is mainly derived from the assumption that only a small surface area of the waste package is disturbed in the zone 2 waste packages.

4.6.2 Low Temperature Operating Mode

The SR reference repository design includes a 50-year ventilated preclosure period, a lineal loading of 1.45kW/m, and a constant drift to drift spacing of 81 meters. The reference design would allow boiling to occur several meters into the host rock surrounding the emplacement drifts for tens to hundreds of years after closure, depending on the relative location of the drift (i.e., edge versus center of the potential repository). Alternatively, compensation for some key uncertainties may be achieved by operating the repository differently. Keeping temperatures below the boiling point of water may reduce uncertainties associated with assessing the thermal response of the host geology and its impact on the performance of the repository. Keeping temperatures below 85°C (185°F) and/or relative humidity below 50 percent may reduce the susceptibility of waste packages, and other components of the engineered barrier system, to corrosion.

Several design variables have a significant impact on the temperature and/or relative humidity in the repository. These variables include: (1) the thermal output of the waste packages, (2) the linear power density at which emplacement drifts are loaded, (3) the distance between emplacement drifts, and (4) the duration and rate at which drifts are ventilated after being loaded. Other parameters (e.g., the thermal properties of the mountain and the infiltration flux) also impact the temperature and humidity of the repository. These latter parameters, however, cannot readily be altered by the design of the repository.

In this section, a low temperature operating mode that includes 0.90 kW/m (a change in the linear power density at which emplacement drifts are loaded) and a 100 year ventilated preclosure period (a change in the ventilation duration) is compared to the reference design. This alternate design would not produce boiling in the host rock and would keep waste package temperatures below boiling as well. Note that is just one example of a possible lower temperature operating mode. The variables described above can be combined in many ways to achieve a lower temperature operating mode depending on what thermal goals and design constraints must be satisfied.

The thermal hydrologic results used to represent the low thermal load design in this TSPA comparison are described in *Thermal Hydrology EBS Design Sensitivity Analysis* (CRWMS M&O 2000 [152201]). To facilitate the comparison between the reference and low thermal load designs, the following key assumptions were made in the analysis:

1. It is assumed that the line-loaded, two-dimensional, drift-scale, thermal-hydrologic submodel, L4C4, that was extracted from the Multiscale Thermal-hydrologic Model (CRWMS M&O 2000 [152201]) is representative of an average location in the potential repository footprint. This assumption is based on the selected submodel's physical location relative to the geometric center of the potential repository footprint (Easting: 170,500m, Northing: 233,800m).

2. Spatial variability in potential repository thermal-hydrologic variables does not have a significant effect on EBS performance. Therefore, thermal-hydrologic output variables from the line-loaded thermal-hydrologic submodel L4C4 can be applied throughout the potential repository.

3. The reduction of power output from 1.45 kW/m to 0.90 kW/m is accomplished by increasing the waste package to waste package spacing. This increase in waste package spacing is accounted for in the two-dimensional model by applying a scaling factor to the original reference design model's thermal power curve. A scaling factor of 0.621 (0.90/1.45 = 0.621) was used (CRWMS M&O 2000 [152201]).

4. The reduction in lineal power output is assumed to be accomplished by increasing waste package to waste package spacing in the emplacement drifts. As a result the potential repository footprint should also increase accordingly. In the analyses presented here, it is assumed that effects of an increased footprint on potential repository performance are not important and can be neglected.

5. The effects and uncertainties associated with coupled thermal-hydrologic-chemical-mechanical processes, such as dissolution/precipitation of minerals and thermally induced fracturing in the host rock, may decrease in magnitude as the potential repository thermal loading decreases. These potential decreases in effects and uncertainties are neglected in the present comparison between the reference design case and the low thermal load case.

The thermal hydrologic results comparing the low thermal load case with the reference design are illustrated in Figures 4.6-7 and 4.6-8. The waste package surface temperature comparisons in

Figure 4.6-7 show that surface temperatures in the low thermal load case reach a much lower peak temperature as expected. As discussed previously in Section 4.6.1, an increase in waste package surface temperature is accompanied by a corresponding decrease in relative humidity around the waste package. As shown in Figure 4.6-8, this decrease in relative humidity does not occur in the low thermal load case since surface temperatures do not rise significantly.

The initial waste package failure curves are shown in Figure 4.6-9 for the reference design case and the low thermal load case. These curves show that there is not a significant difference in the two cases as they are currently modeled. This result illustrates the insensitivity of the SR waste package corrosion and degradation model to thermal hydrologic conditions around the waste package. Furthermore, since waste package failure does not occur until after 10,000 years when both cases exhibit similar thermal hydrologic conditions, the overall dose comparison between the two cases will not show a significant difference either. This result is shown in Figure 4.6-10.

In conclusion, although this low temperature operating mode reduces waste package surface temperatures and increases the relative humidity around waste packages these effects do not significantly impact waste package performance. In addition, since waste package failure does not occur until after 10,000 years, the thermal hydrologic conditions for both cases are similar during the period when radionuclides are mobilized. As a result, doses for both cases show very little difference. It should be emphasized that these results are based on the simplifying assumptions such as the effects of coupled thermal-hydrologic-chemical-mechanical processes on potential repository performance in both cases may be accounted for by using the same models and input data. Given the assumptions, the example demonstrates that lower temperature operating modes may perform in a manner comparable to high temperature operating modes. Other combinations of: (1) the thermal output of the waste packages, (2) the linear power density at which emplacement drifts are loaded, (3) the distance between emplacement drifts, and (4) the duration and rate at which drifts are ventilated after being loaded may be evaluated to optimize the system depending on design goals and any operating constraints that may be imposed. Model refinements may provide insight into the potential for reducing uncertainties that are not currently available based on the simplifying assumptions that were used in this particular analysis.

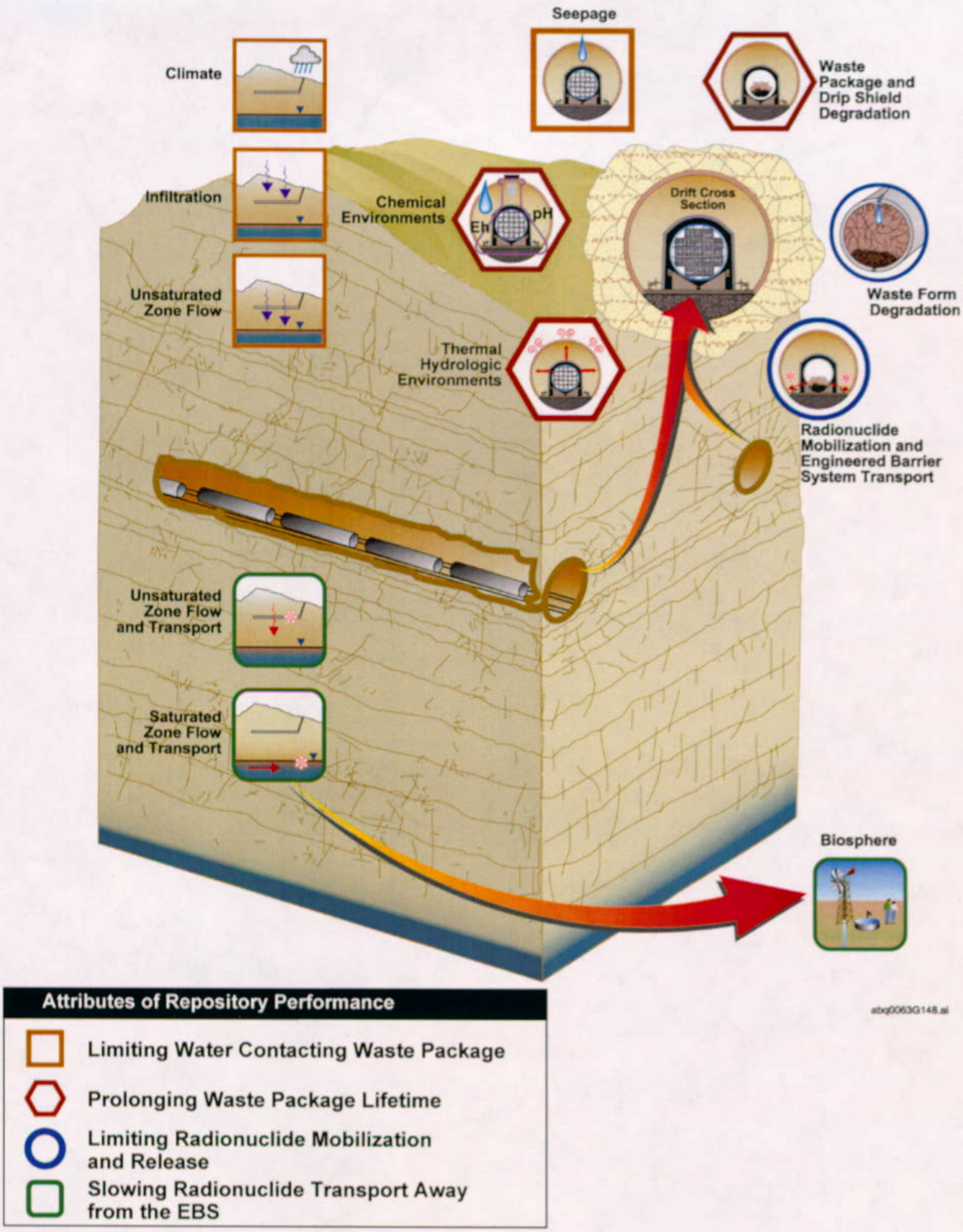


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Figure 4.1-1. Illustration Showing the Important Total System Performance Assessment Submodels for the Nominal Scenario

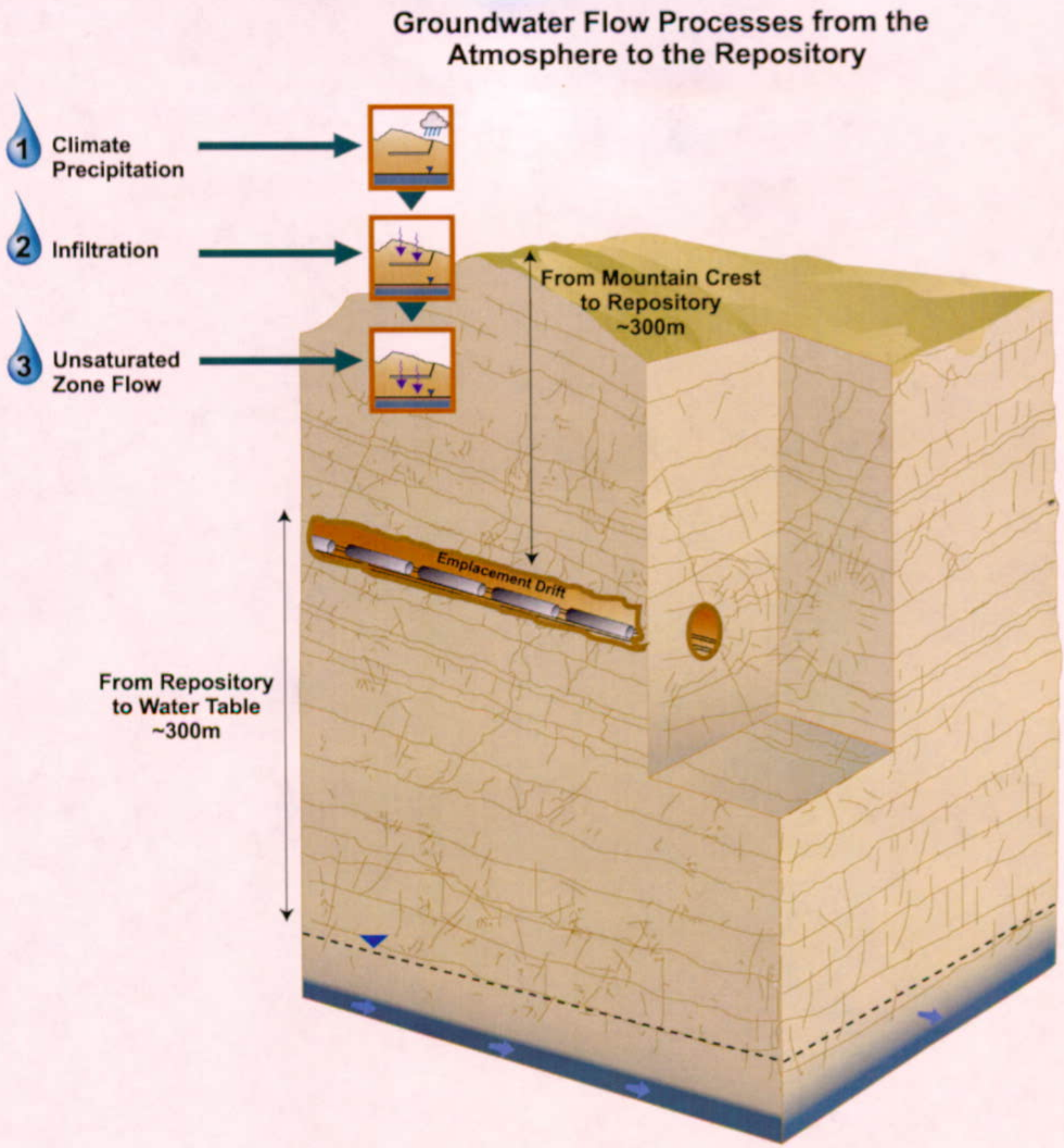

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Figure 4.1-2. Total System Performance Assessment Submodels for Groundwater Flow above the Potential Repository

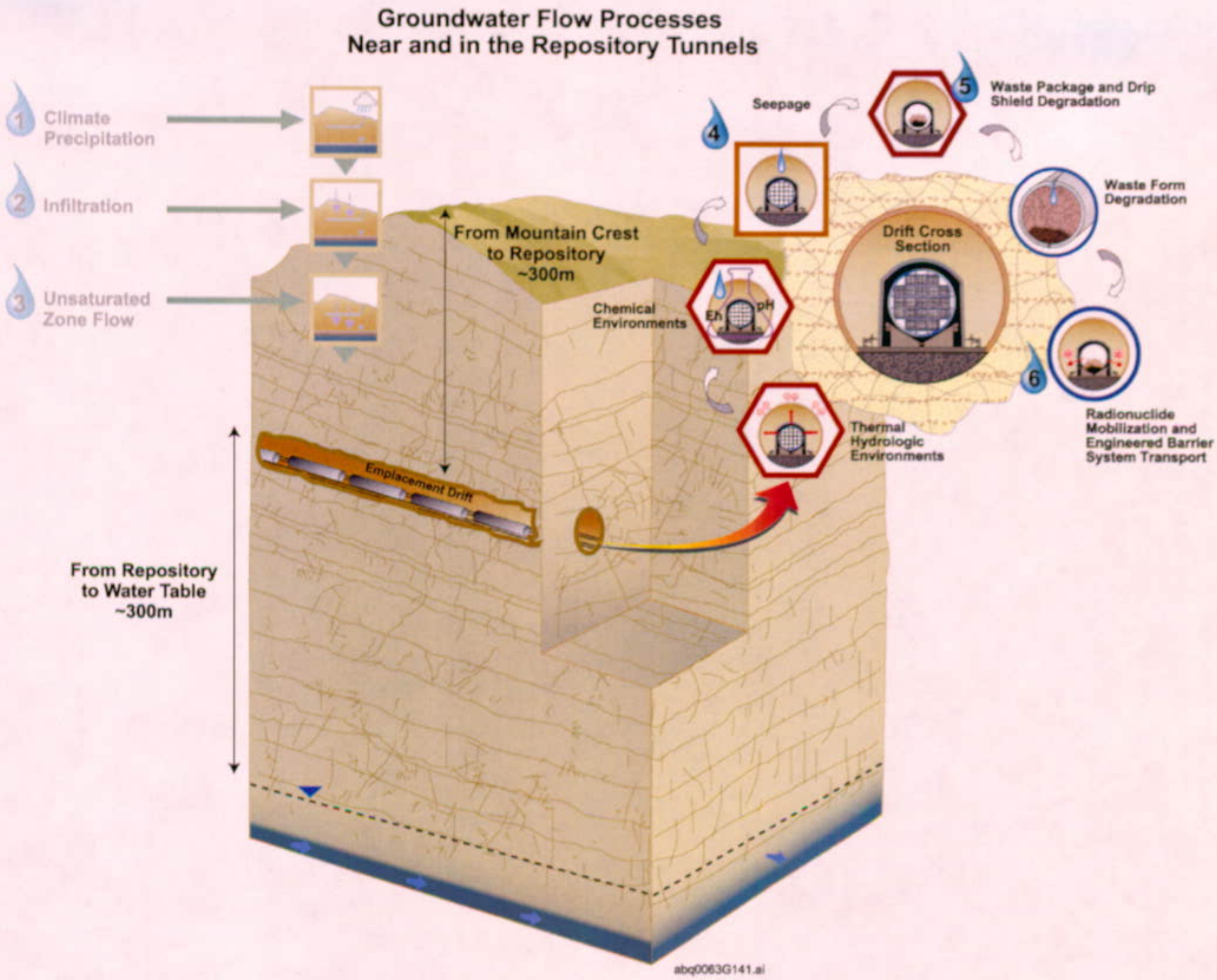

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Figure 4.1-3. Total System Performance Assessment Submodels for Flow and Transport near and in the Potential Repository Tunnels

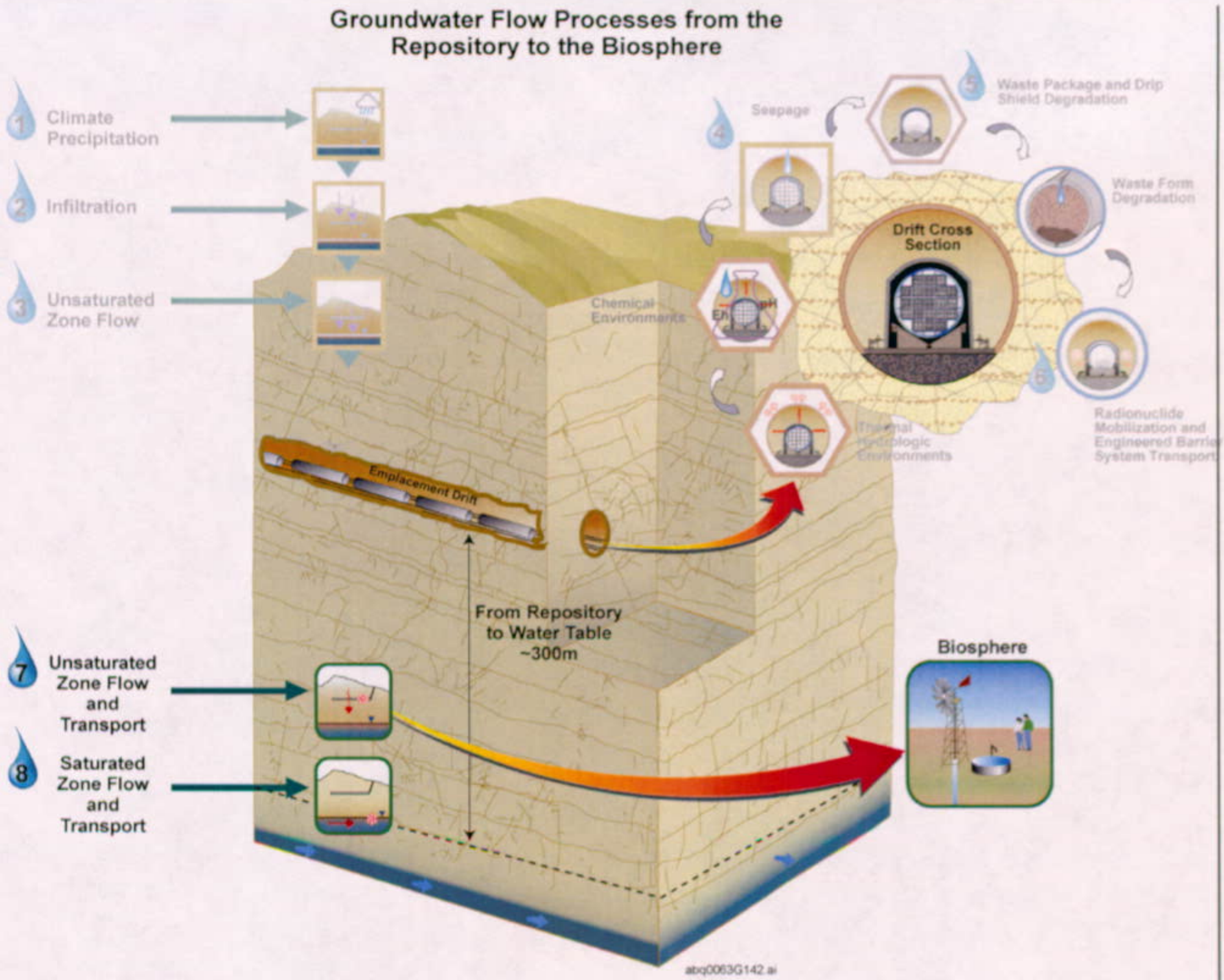

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Figure 4.1-4. Total System Performance Assessment Submodels for Flow and Transport from the Potential Repository to the Biosphere

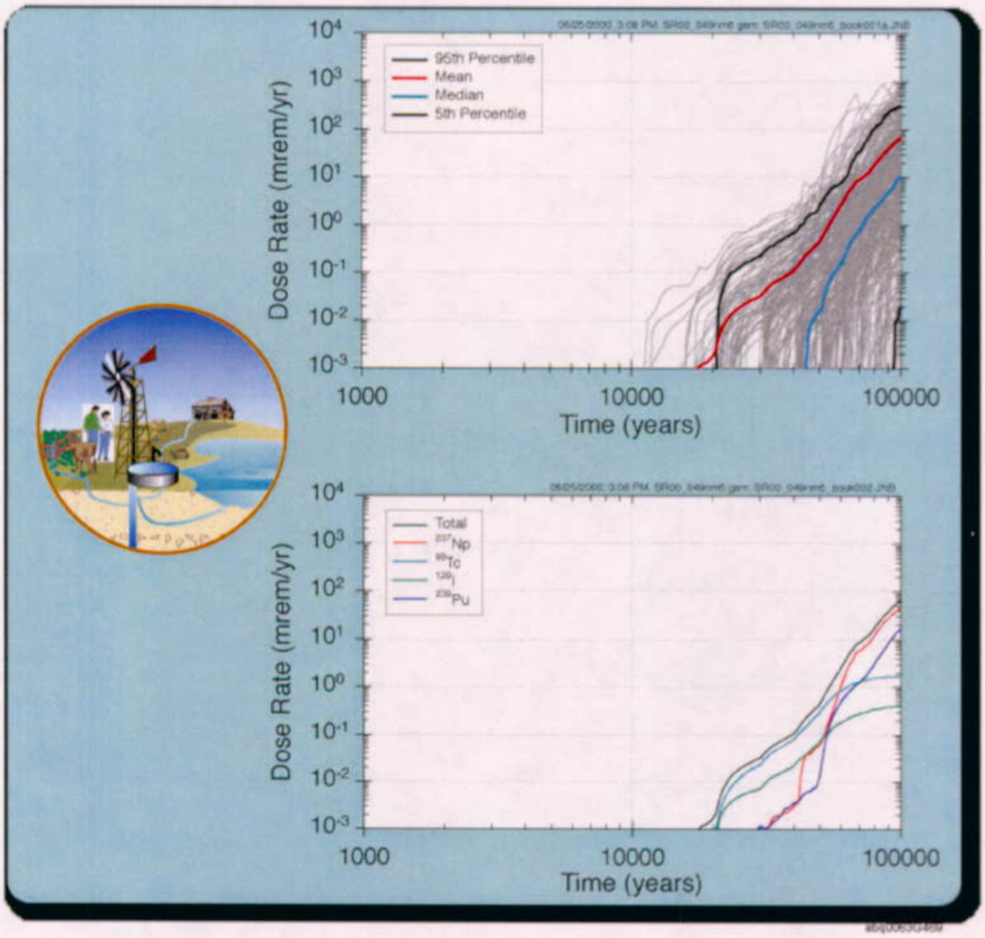

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Figure 4.1-5. Simulated Annual Dose Histories for the Nominal Scenario

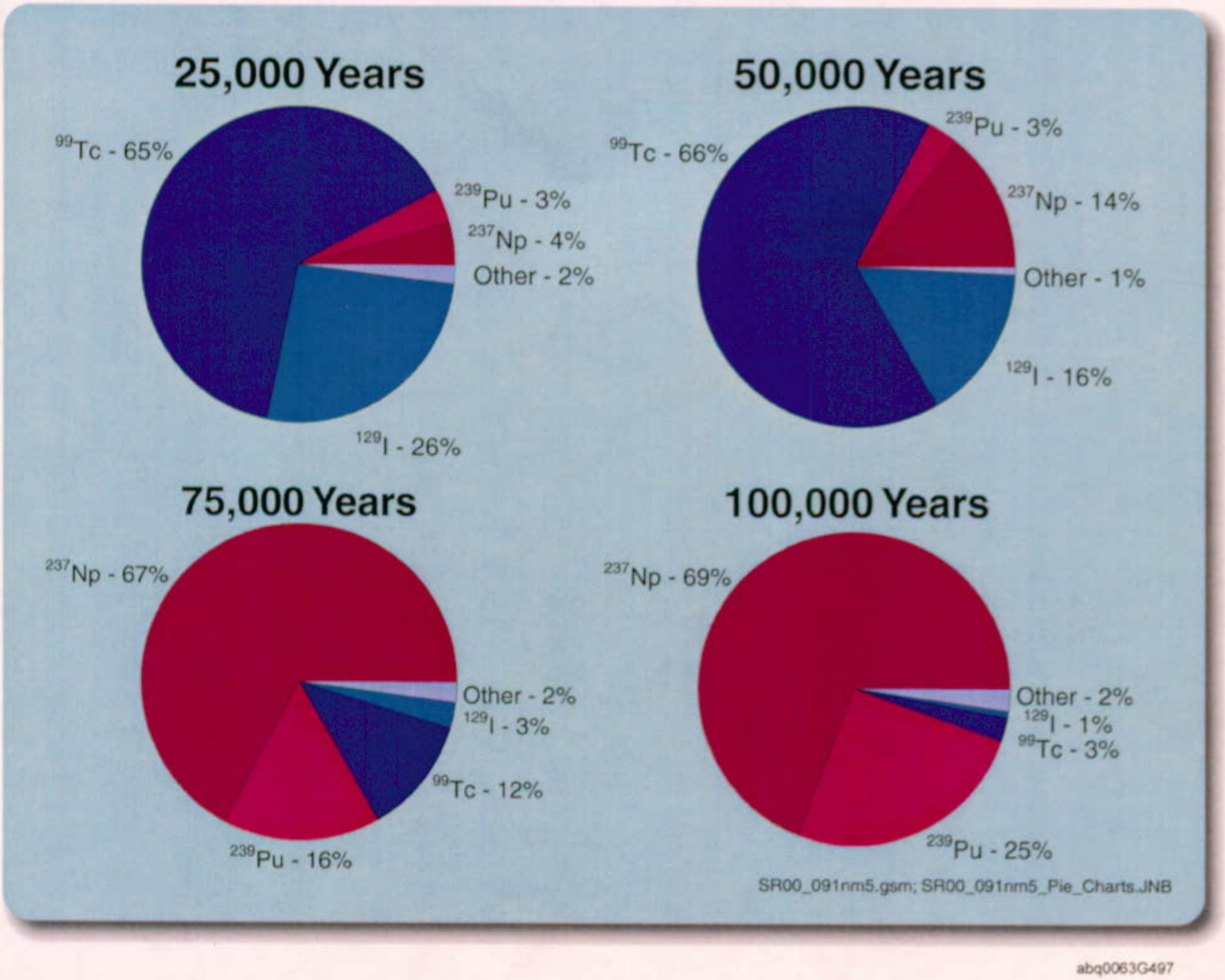

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Figure 4.1-6. Contribution of Radionuclides to the Mean Annual Dose at Four Times

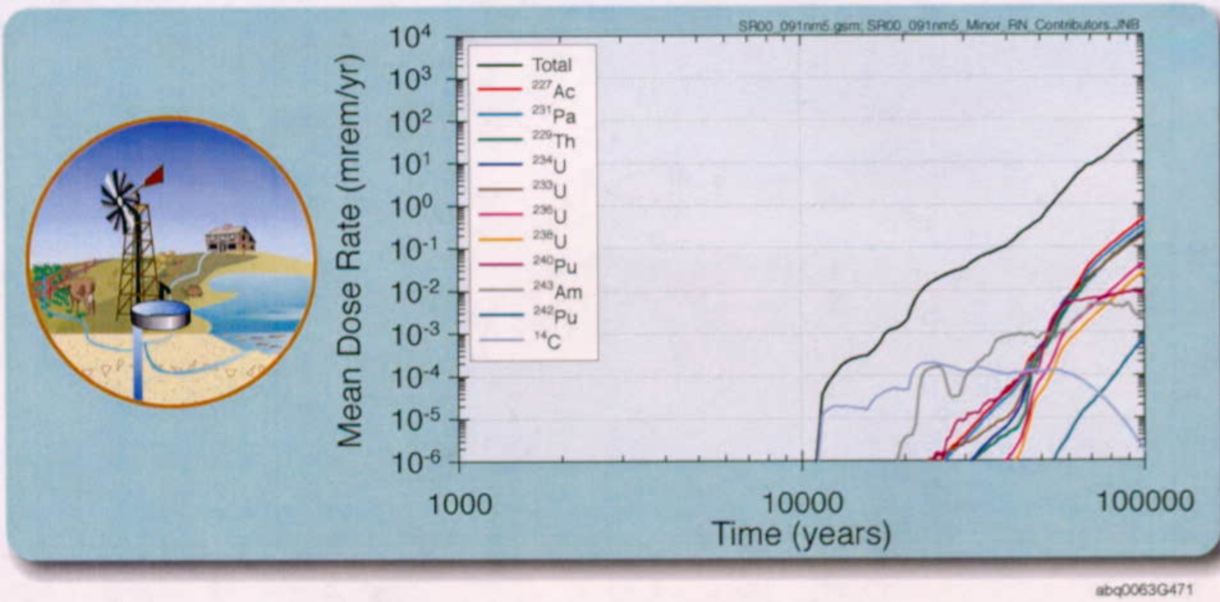

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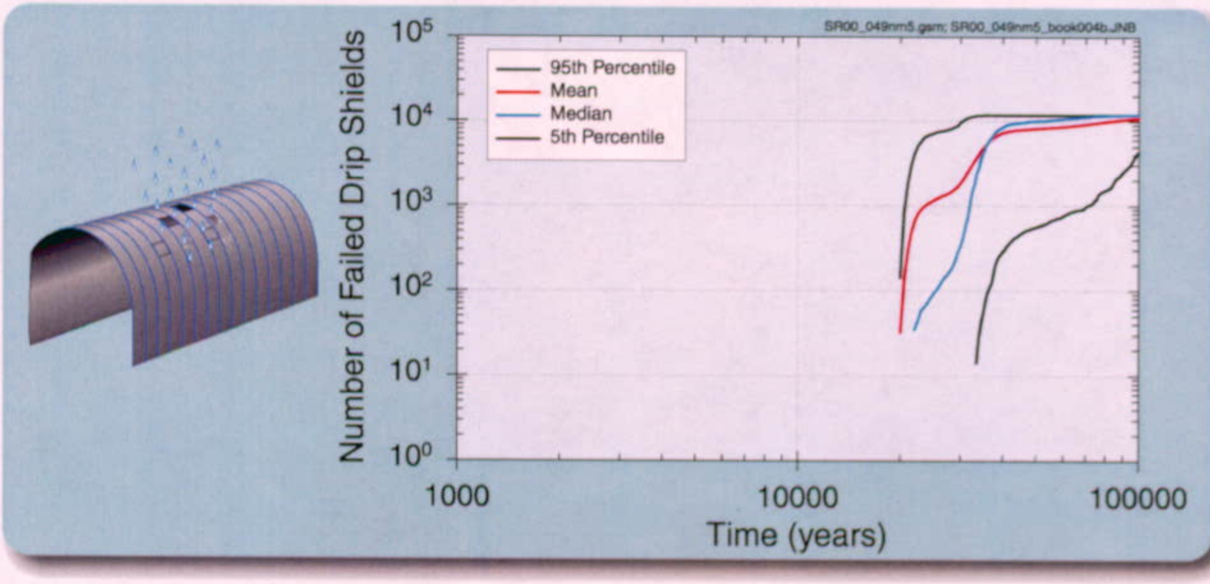
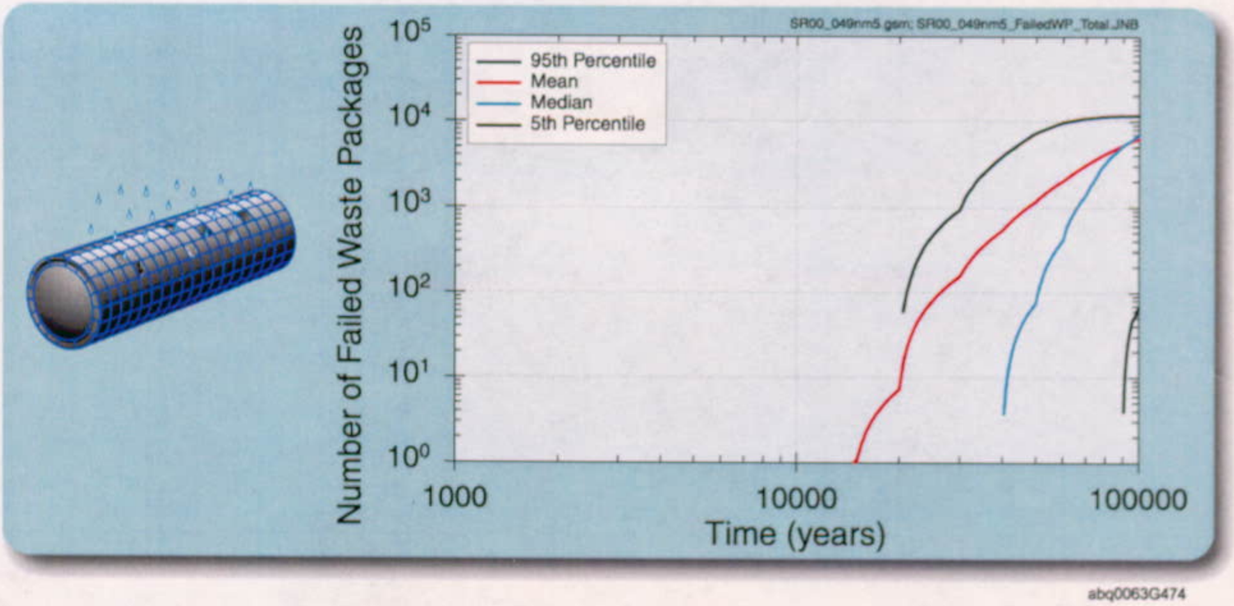
Figure 4.1-7. Mean Annual Dose Histories for Less-Important Radionuclides



abq0063G473

Figure 4.1-8. Number of Failed Drip Shields over Time



abq0063G474

Figure 4.1-9. Number of Failed Waste Packages over Time

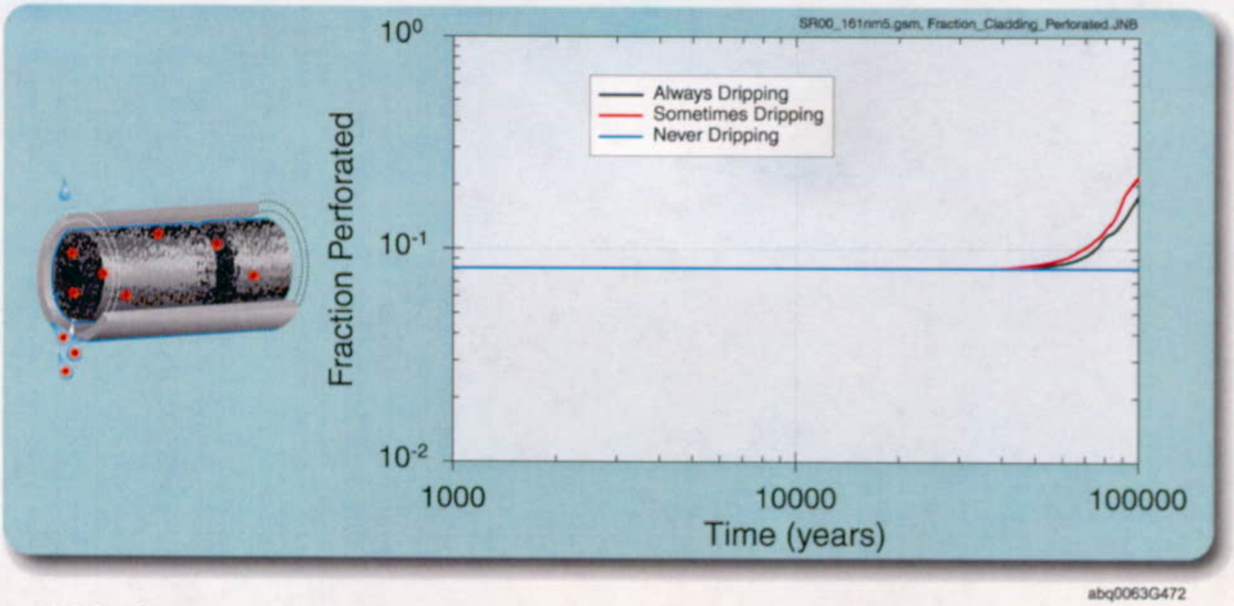


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Figure 4.1-10. Fraction of Failed Cladding over Time

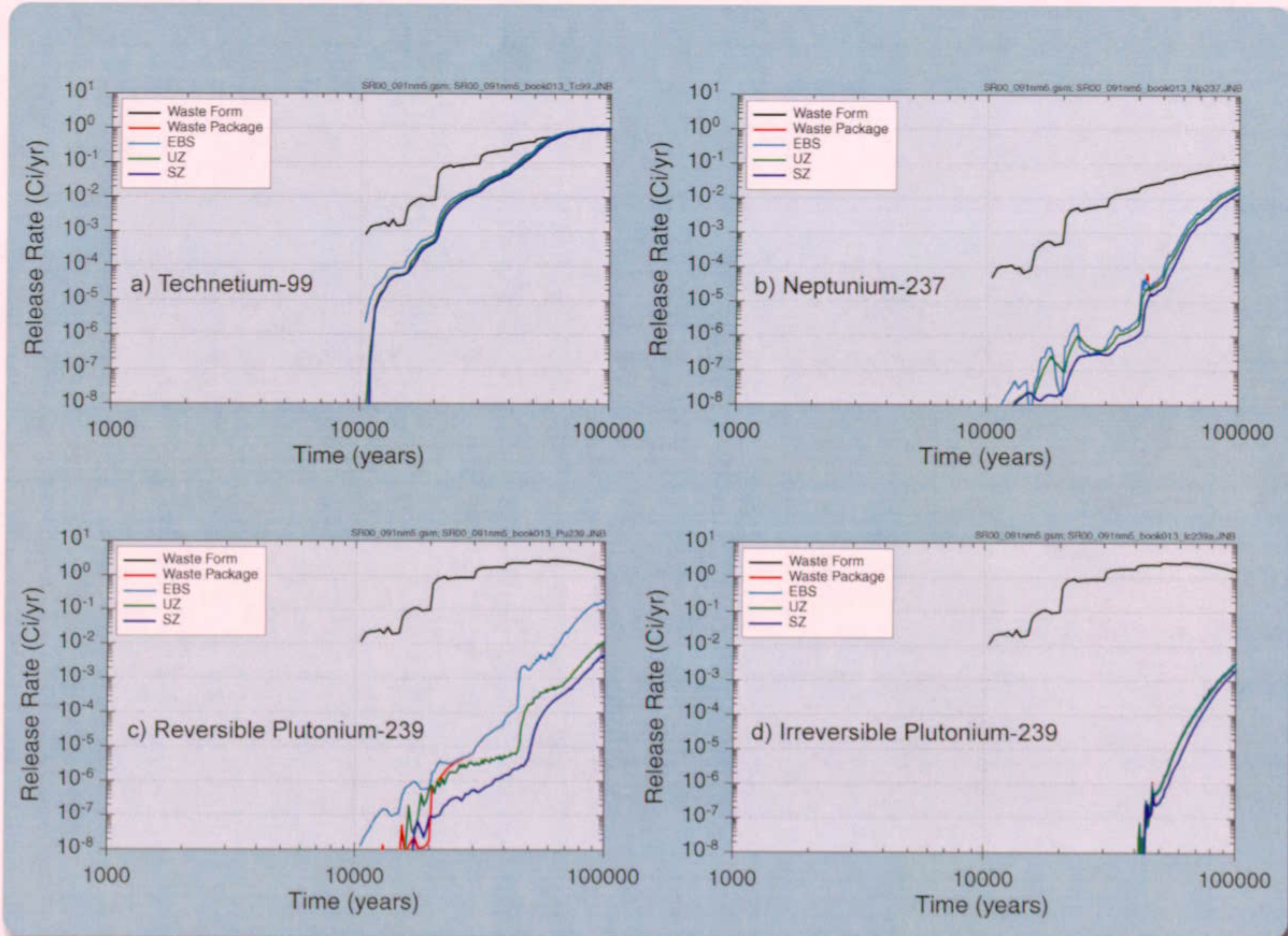

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Figure 4.1-11. Mean Radionuclide Release Rate from Five Locations

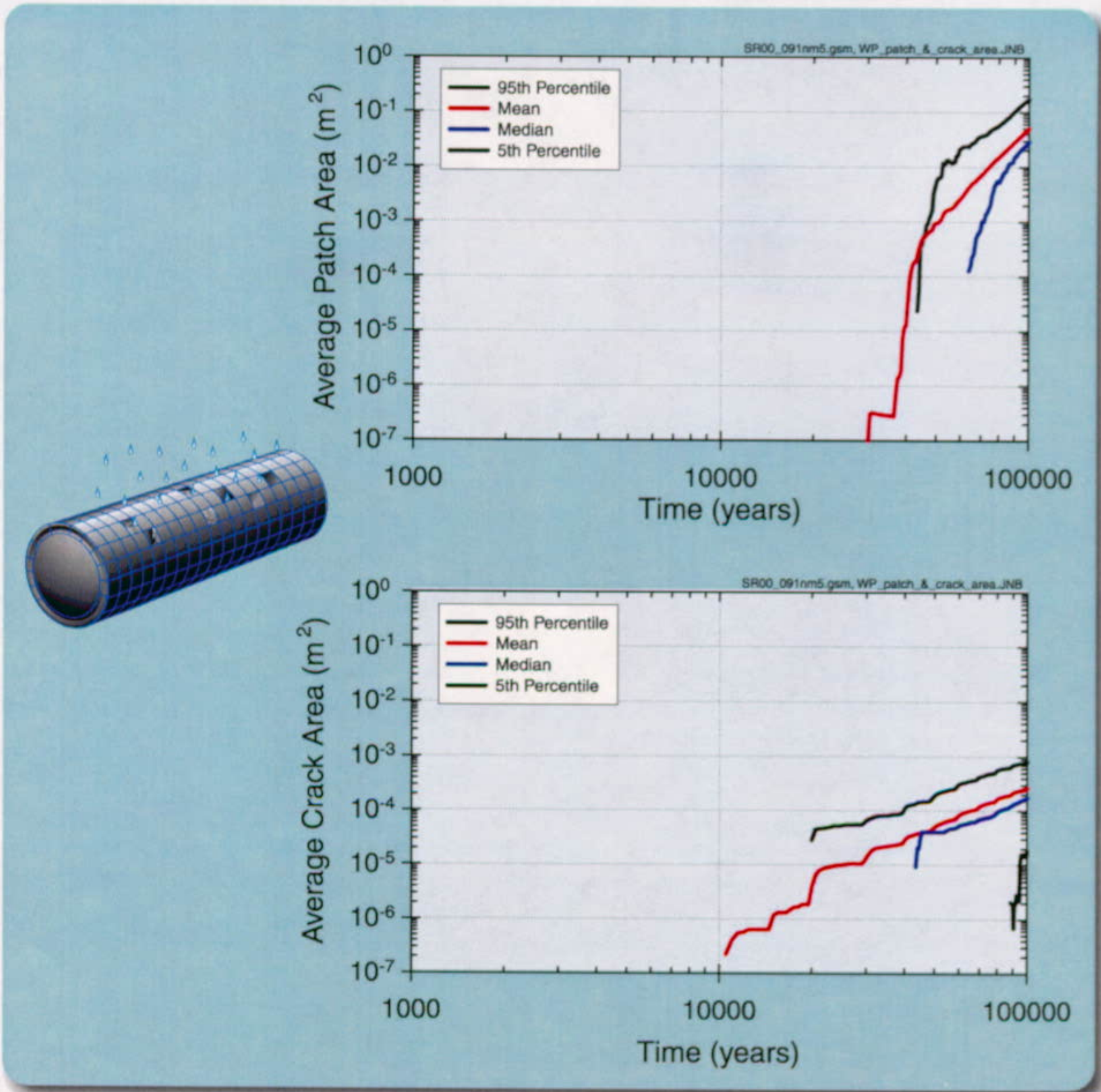

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Figure 4.1-12. Waste Package Opening Area over Time

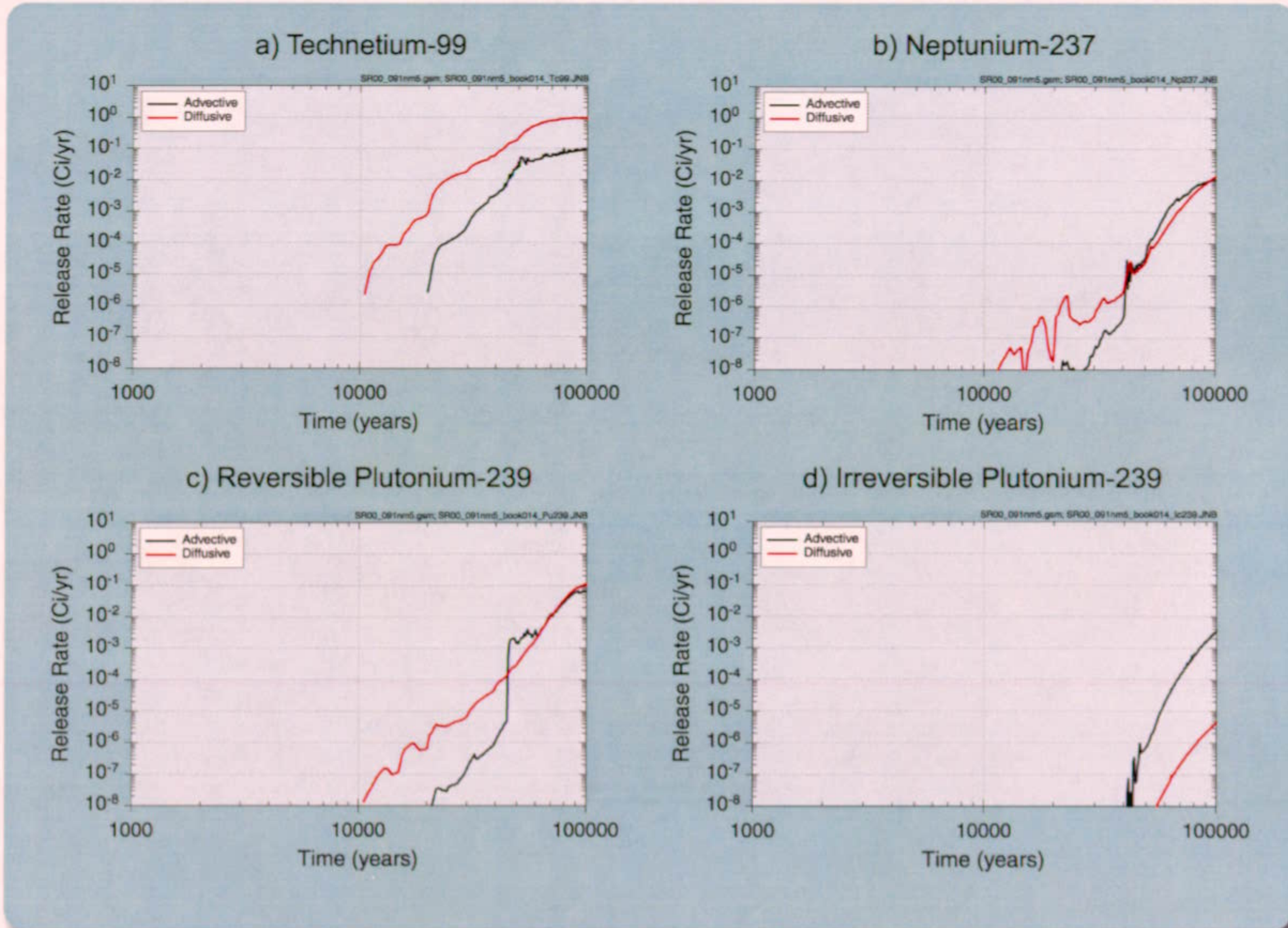

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Figure 4.1-13. Mean Diffusive and Advective Releases from the Engineered Barrier System

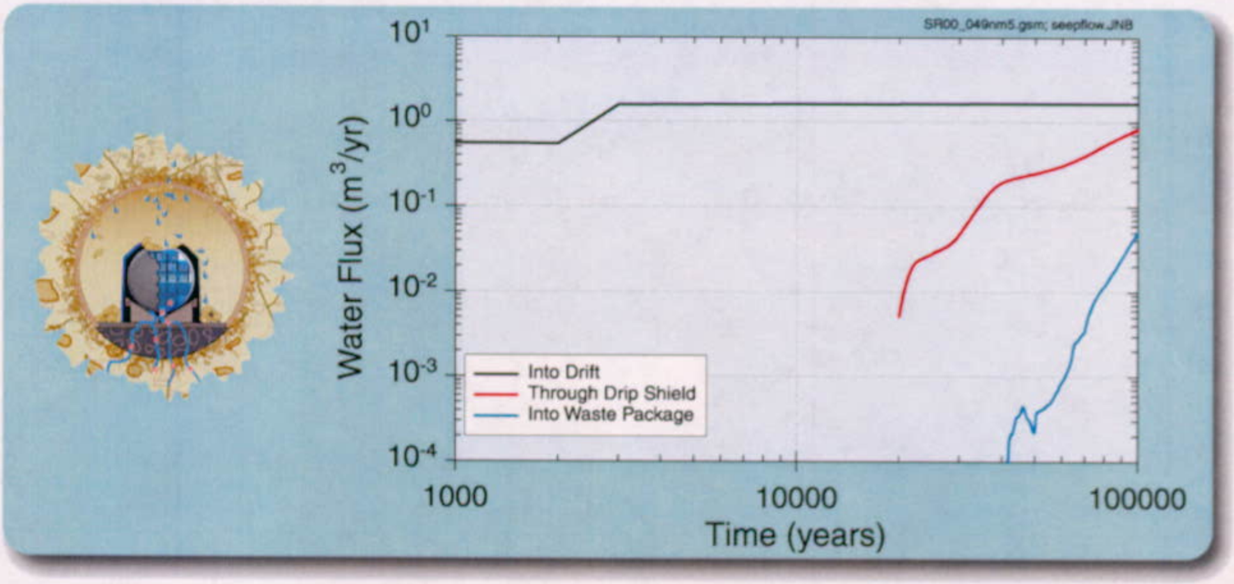

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Figure 4.1-14. Mean Water Flow Rates for Sometimes-Seeping Locations in the 20 to 60 mm/yr Infiltration Bin

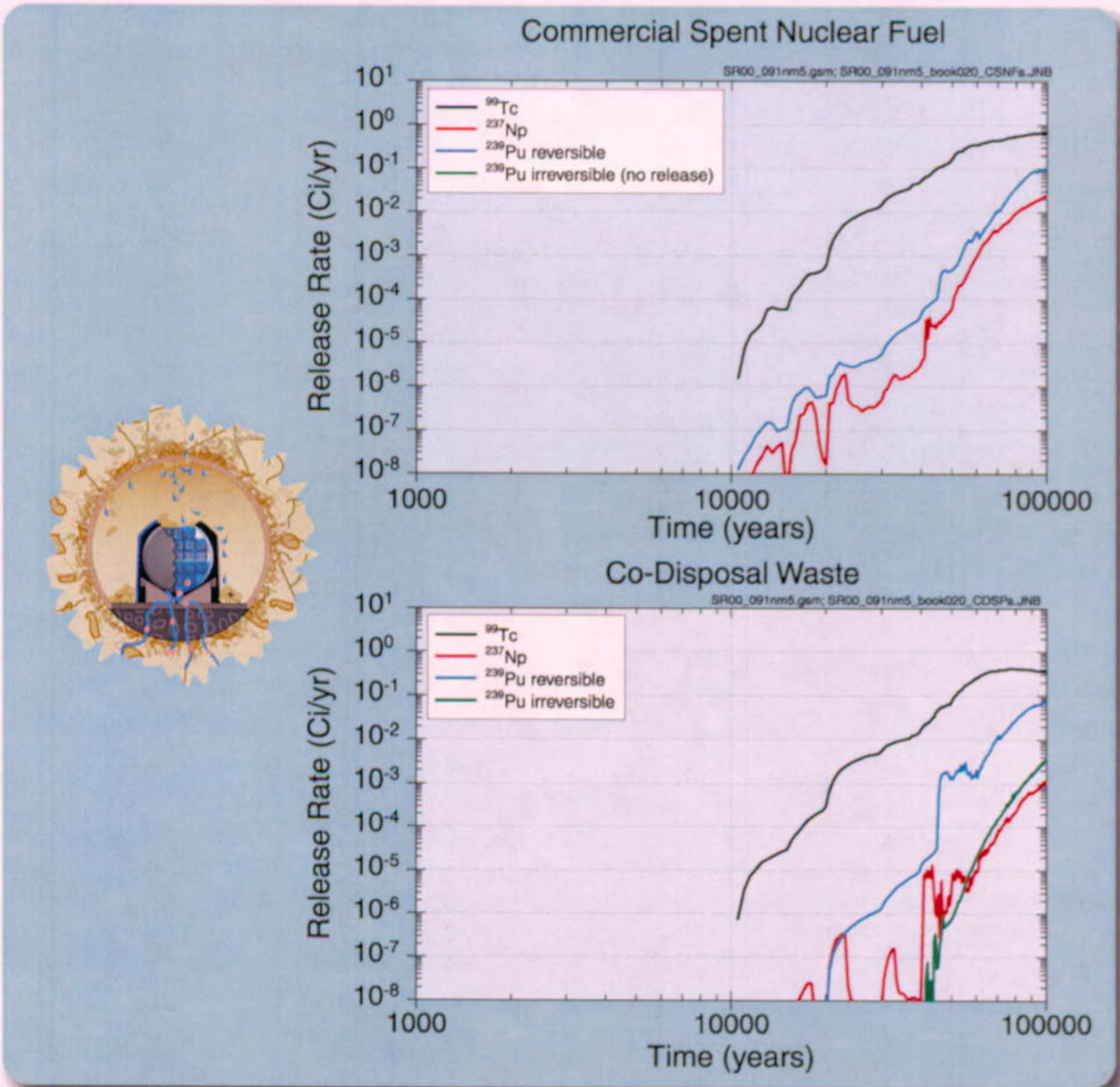

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Figure 4.1-15. Mean Release Rates from the Engineered Barrier System for Two Waste Types

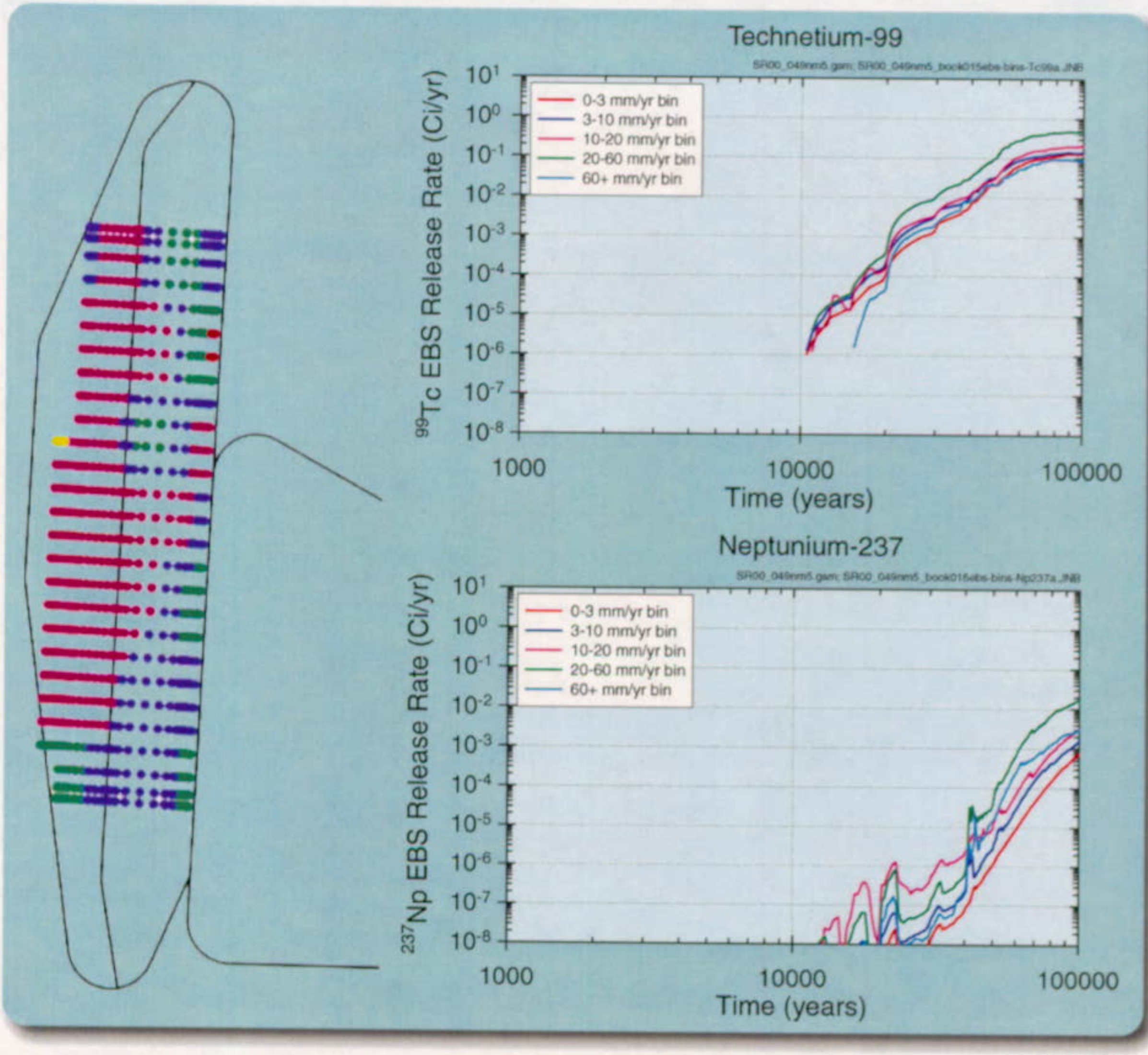

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Figure 4.1-16. Mean Engineered Barrier System Releases from the Five Infiltration Bins

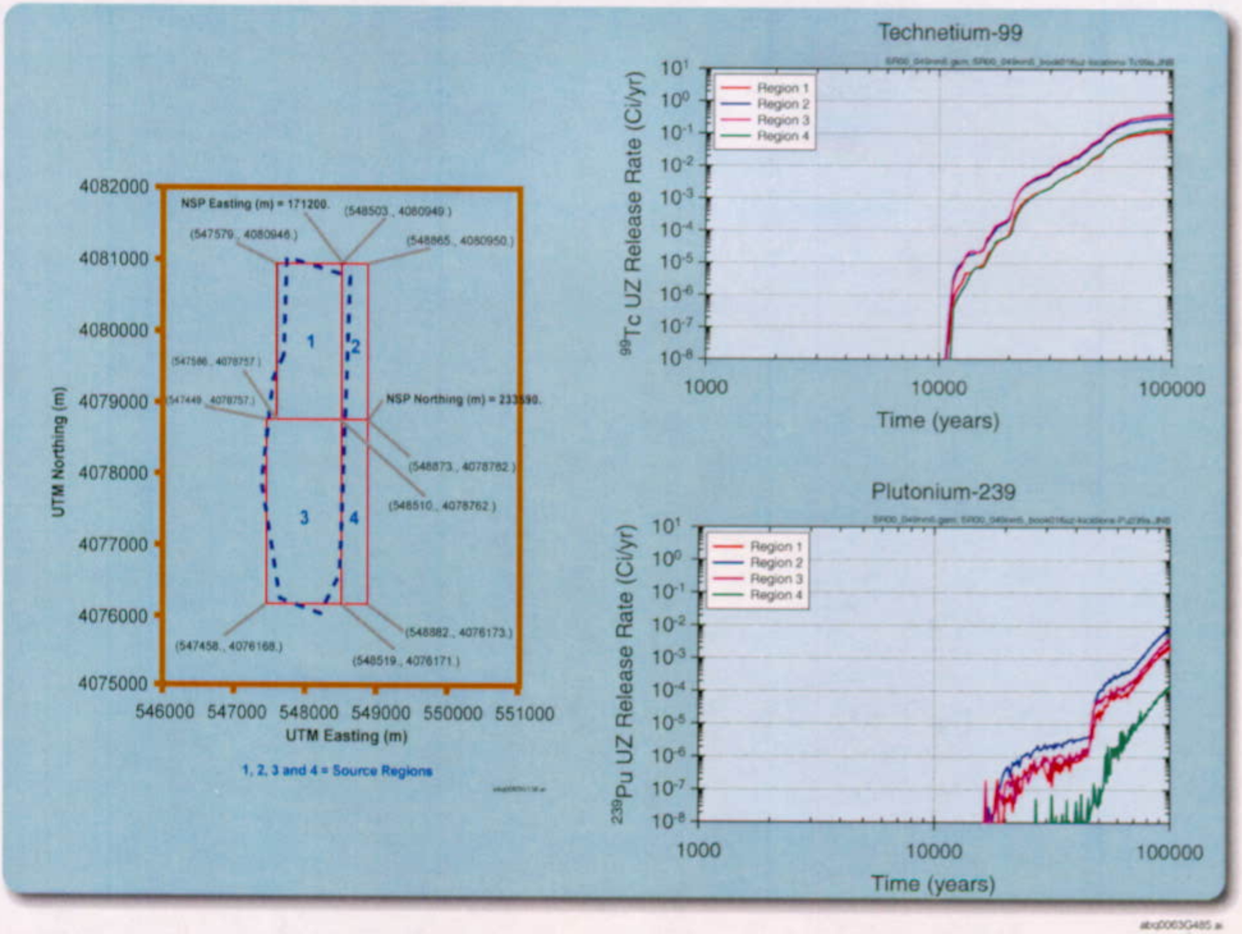
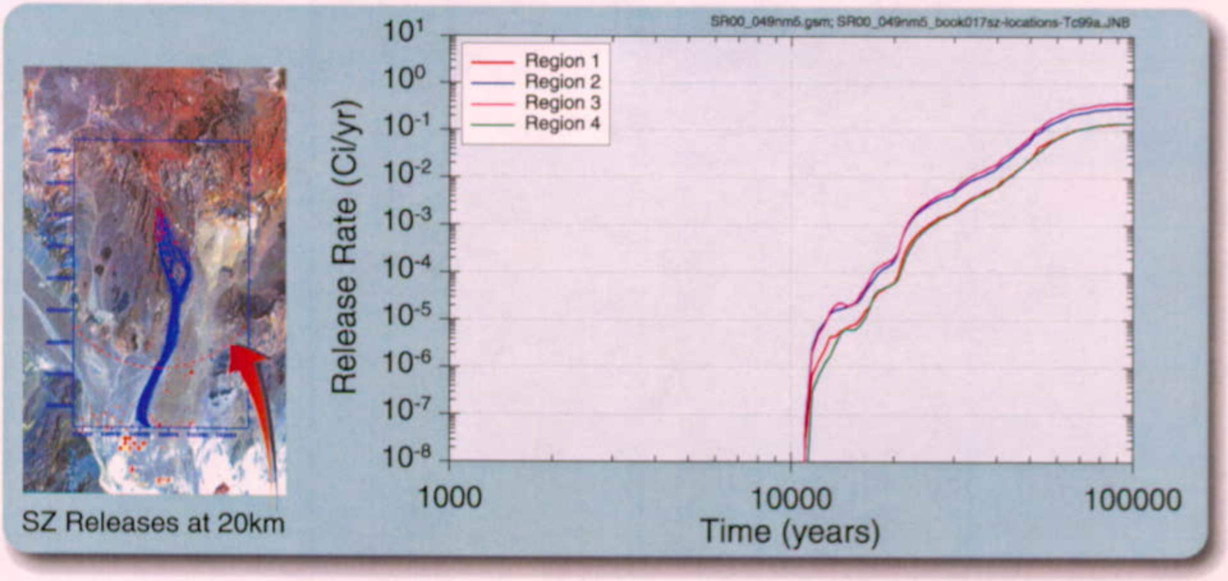


Figure 4.1-17. Mean Unsaturated Zone Releases from Four Regions



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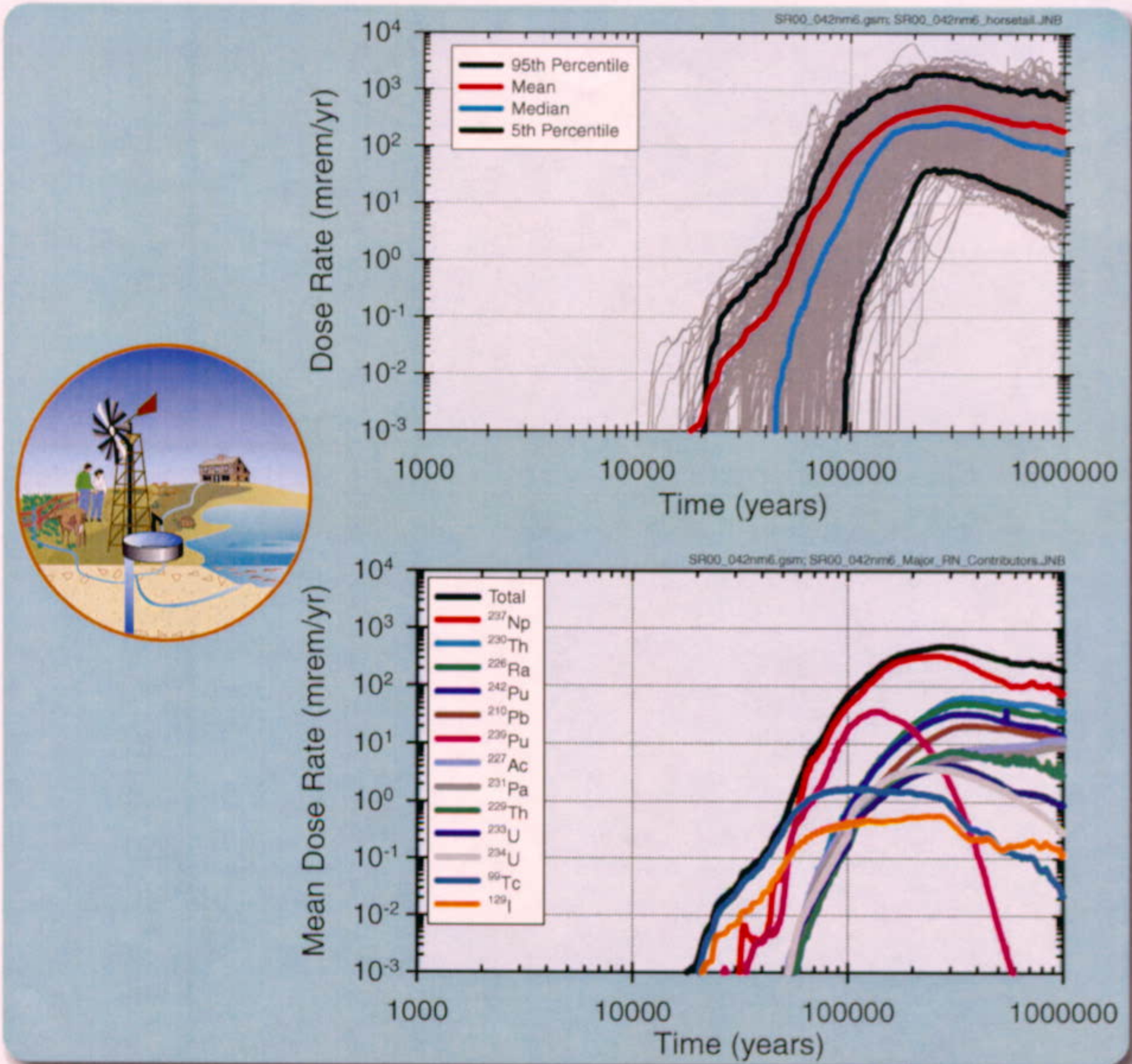
NOTE: SZ graphic is same as Figure 3.8-12

Figure 4.1-18. Mean Saturated Zone Releases of ^{99}Tc for Four Source Regions



abq0063G498

NOTE: Upper plot presents the full distribution of multiple realizations and key statistical representations of this distribution and the lower plot presents the key radionuclides associated with the mean dose rate derived from the multiple realizations.

Figure 4.1-19a. Simulated Annual Dose Histories for the Nominal Scenario over 1,000,000 Years Using the Base Case Models

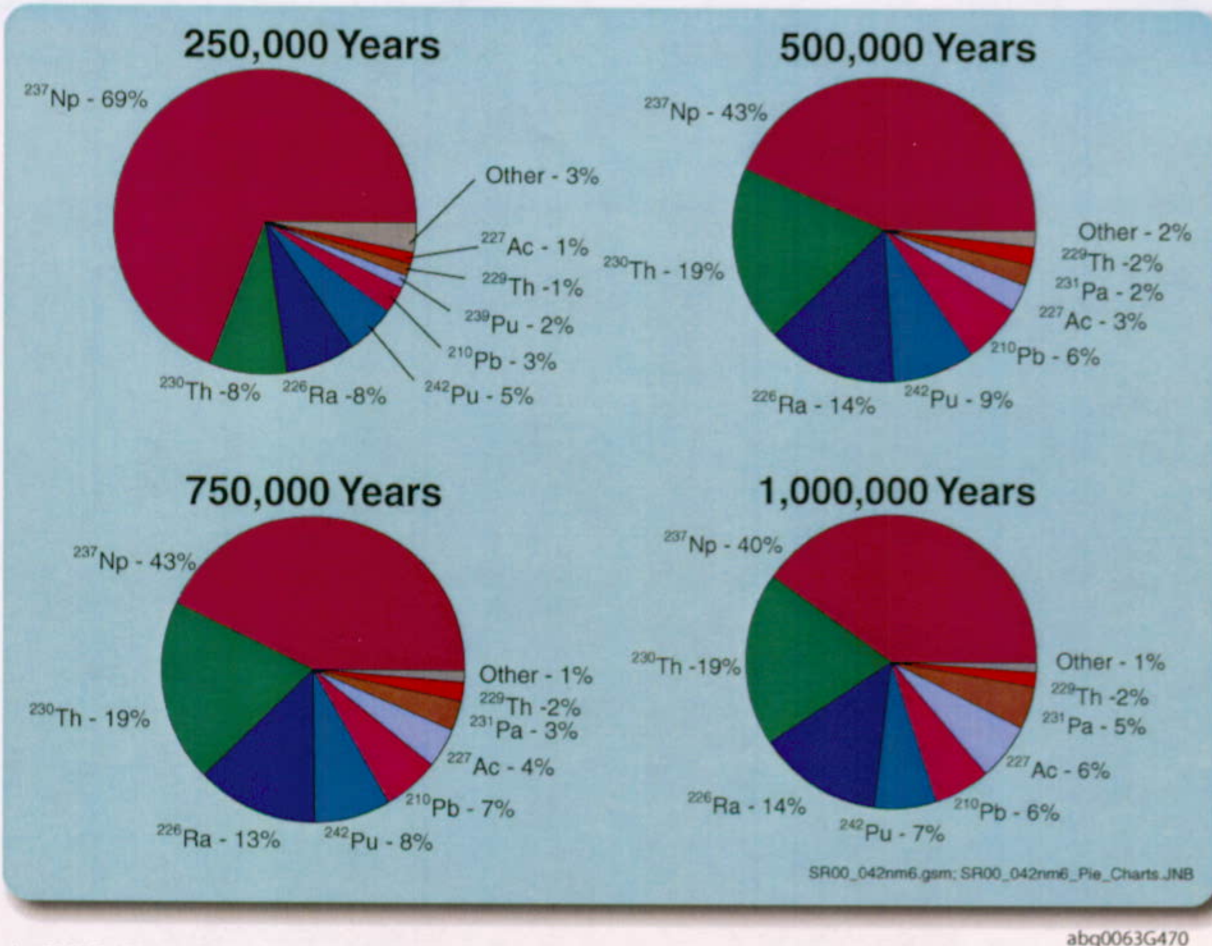


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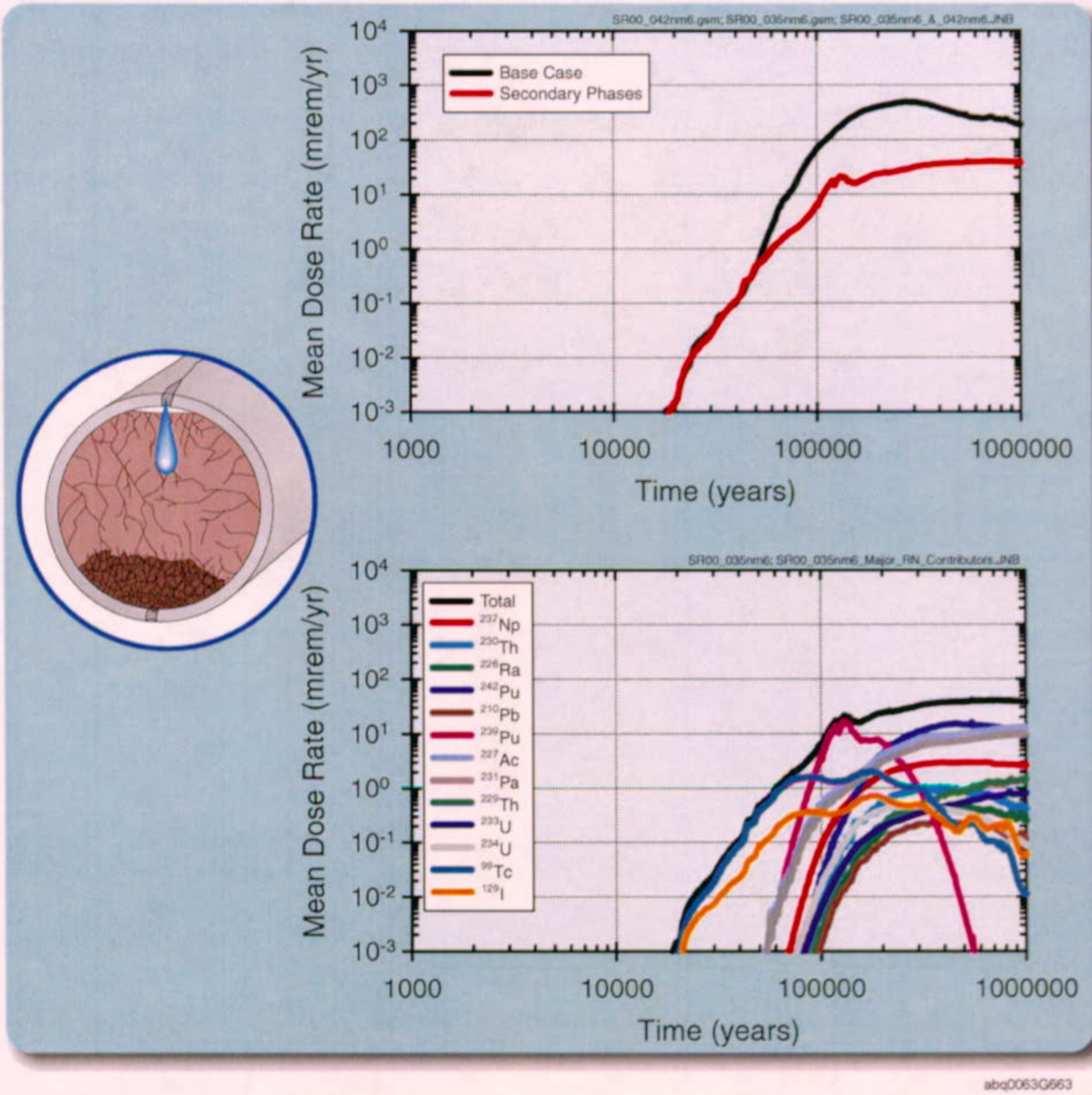
Figure 4.1-19b. Key Radionuclides Affecting Mean Annual Dose for the Nominal Scenario over 1,000,000 Years Using the Base Case Models



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NOTE: Upper plot presents the full distribution of multiple realizations and key statistical representations of this distribution and the lower plot presents the key radionuclides associated with the mean dose rate derived from the multiple realizations.

Figure 4.1-20. Simulated Annual Dose Histories for the Nominal Scenario over 1,000,000 Years Using the Long-Term Secondary Phase Solubility Model

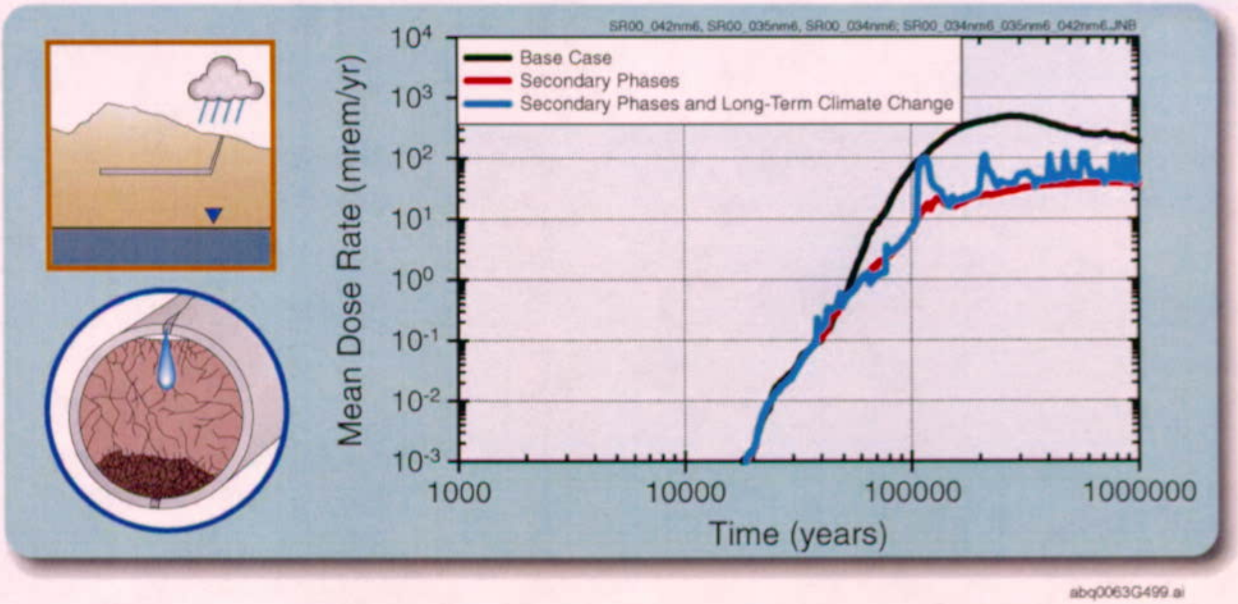

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Figure 4.1-21. Simulated Mean Annual Dose for the Nominal Scenario over 1,000,000 Years Using the Long-Term Secondary Phase Solubility Model and the Post-10,000 Year Climate Model

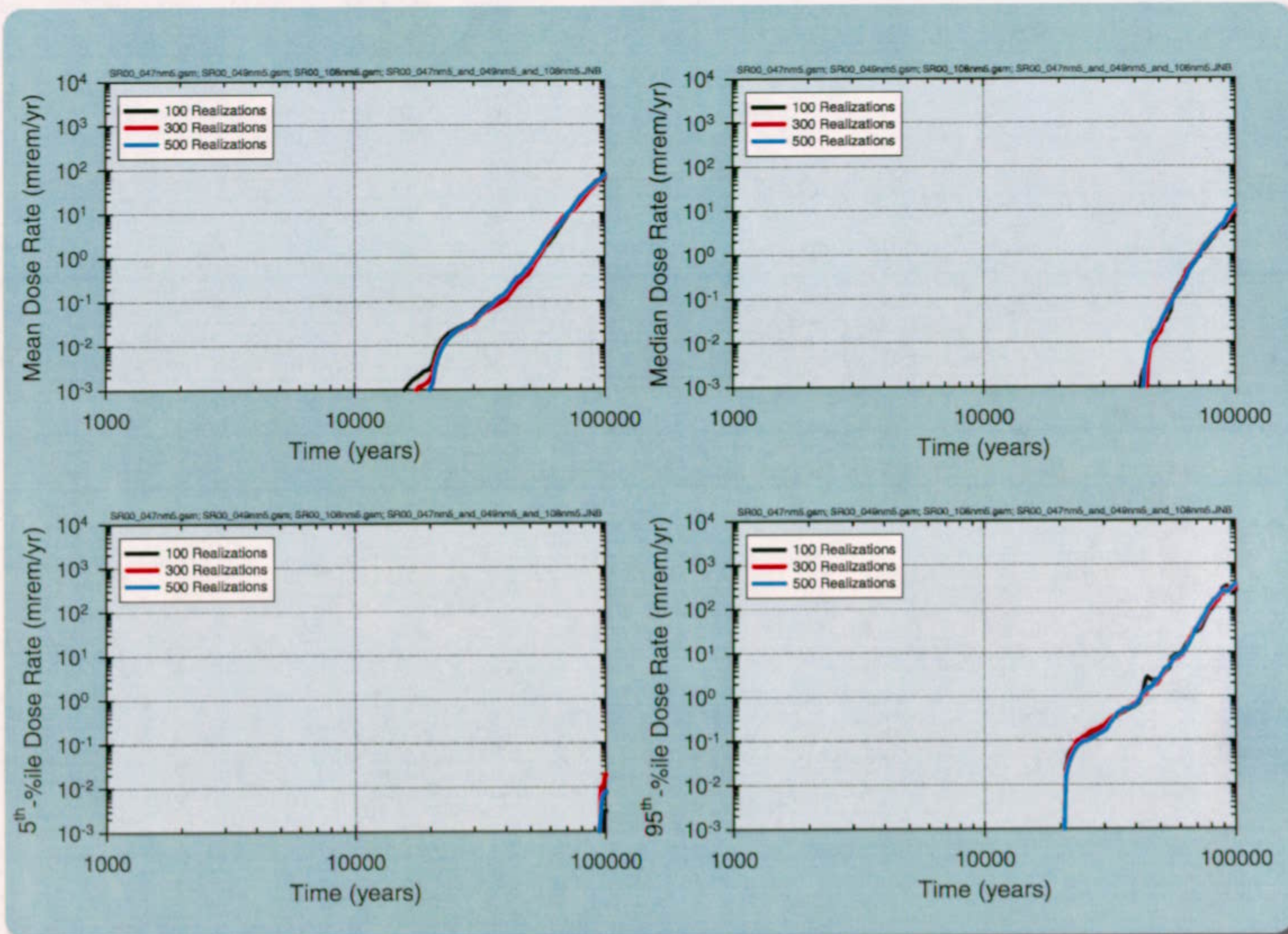

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Figure 4.1-22. Comparison of Annual Dose Results for Different Numbers of Realizations

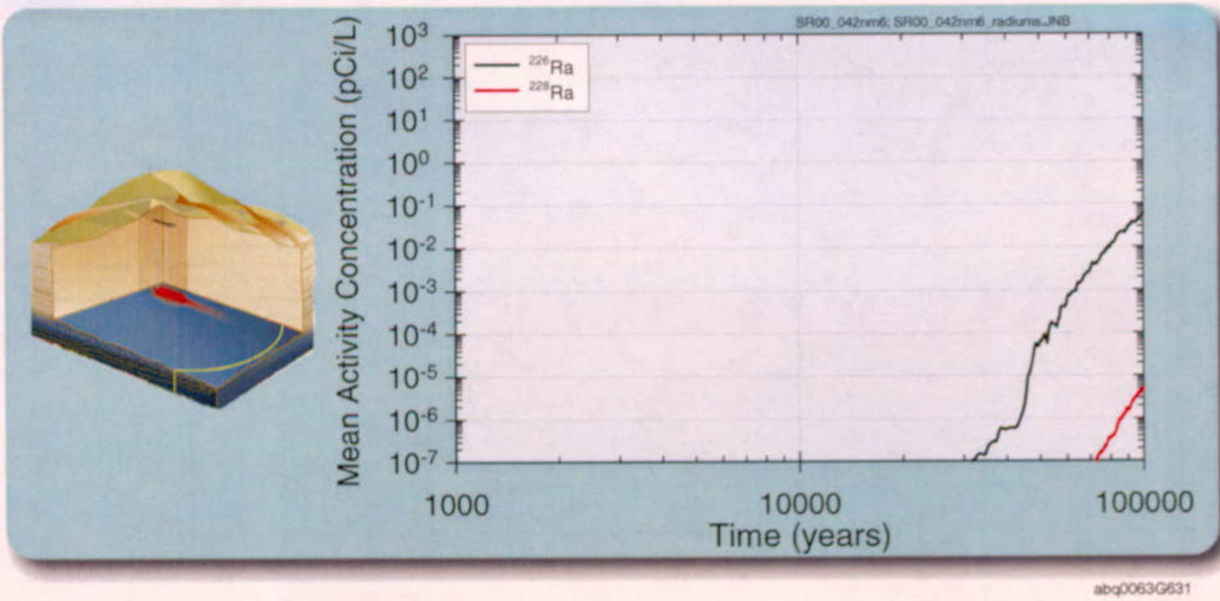

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Figure 4.1-23. Mean Time History of Radium Activity in Groundwater (Excluding Background)

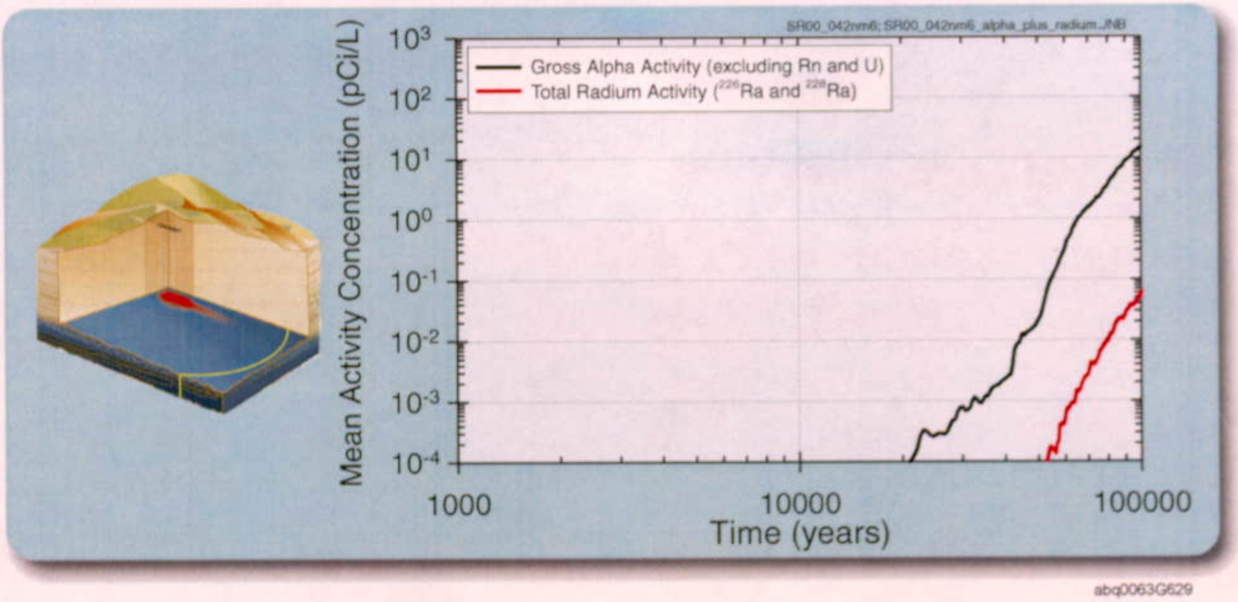

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Figure 4.1-24. Mean Time History of Radium and Gross Alpha Activity in Groundwater (Excluding Background)

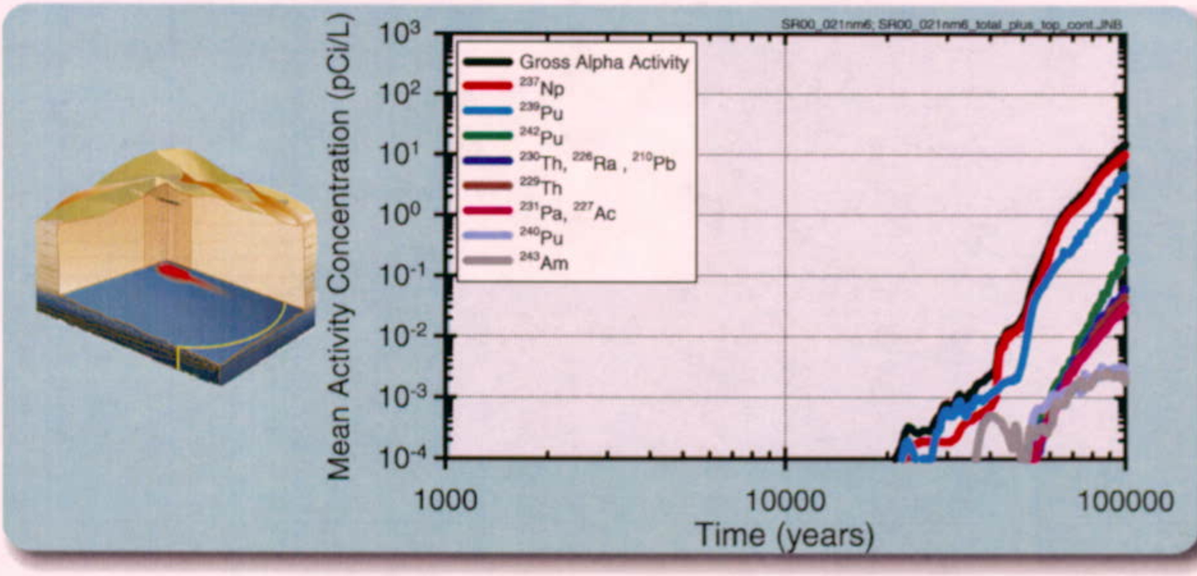

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Figure 4.1-25. Contribution of Radionuclides to the Mean Gross Alpha Activity (Excluding Background)

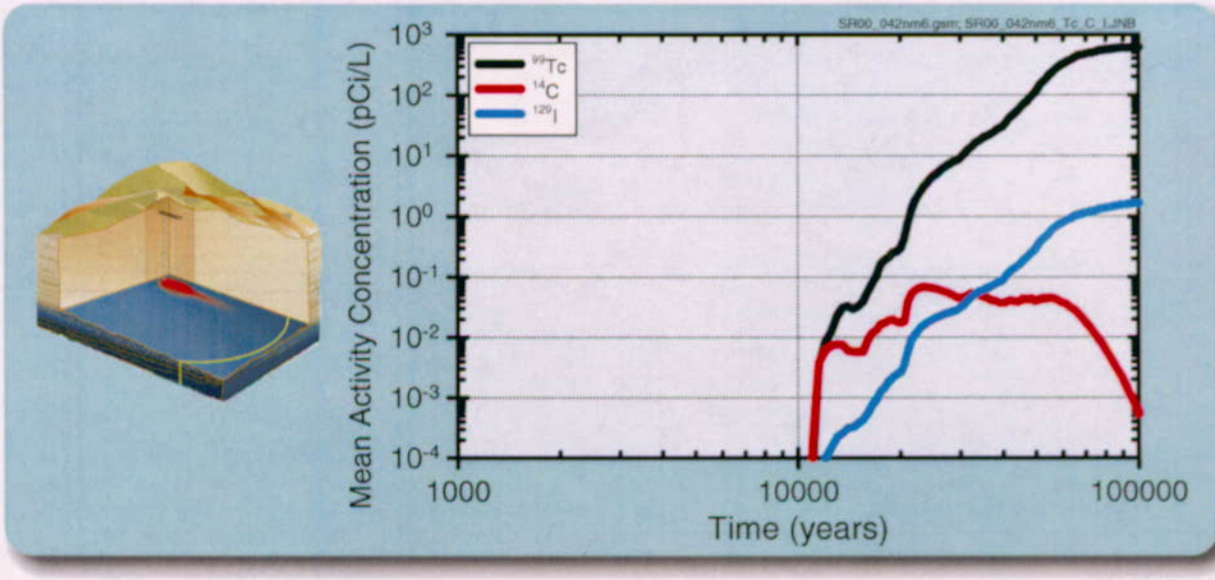

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Figure 4.1-26. Mean Time History of Activity for Beta and Photon Emitters in Groundwater (Excluding Background)

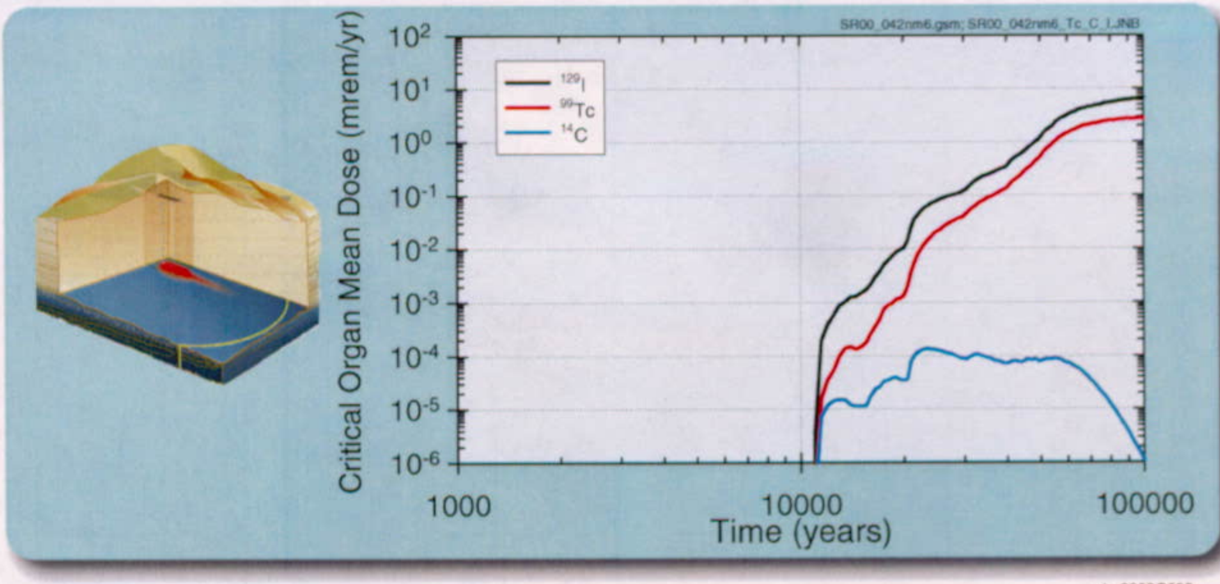

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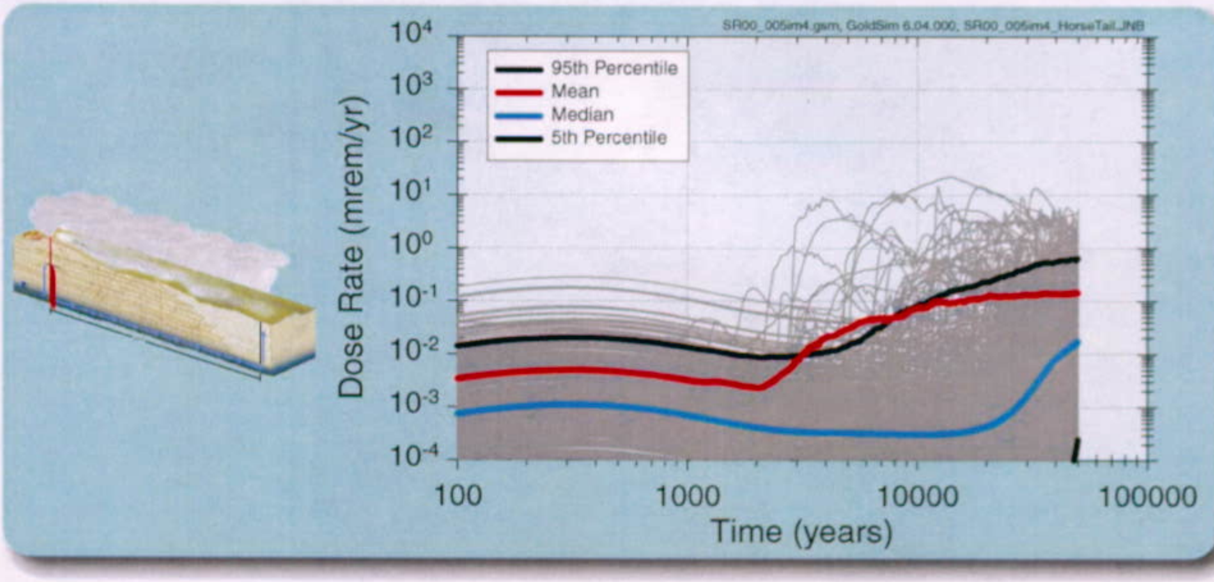
Figure 4.1-27. Mean Time History of Dose to Critical Organ for Beta and Photon Emitters in Groundwater (Excluding Background)



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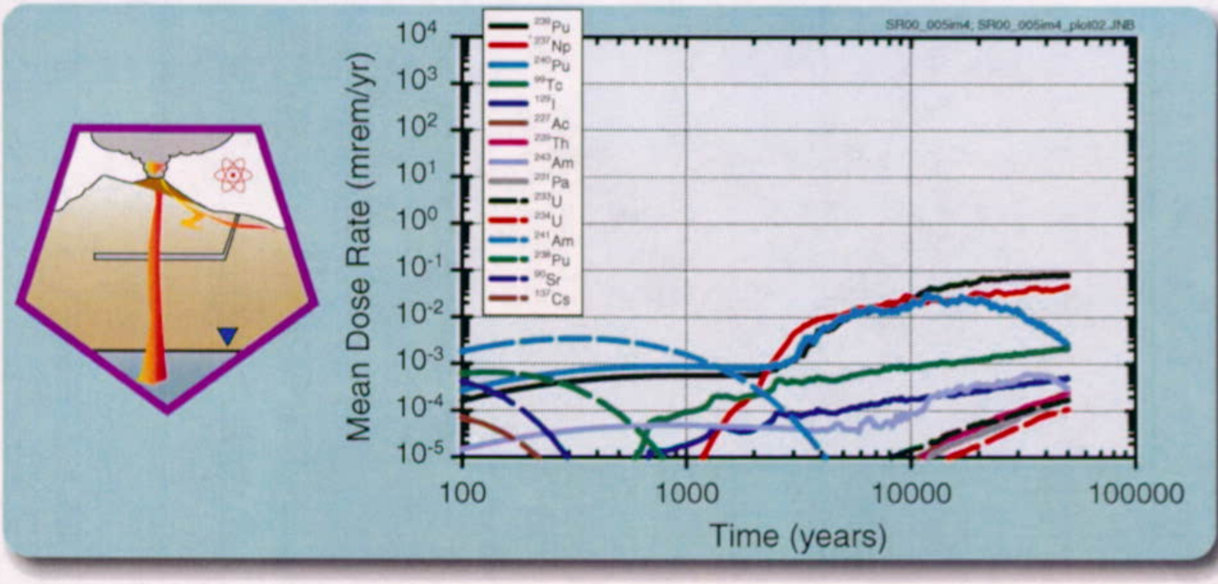
NOTE: Including both eruptive and intrusive releases from the repository.
Simulation was only for 50,000 year time period. See text for discussion.

Figure 4.2-1. Total Probability-Weighted Annual Dose Rate from Igneous Disruption



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NOTE: Igneous disruption shown in Figure 4.2-1.

Figure 4.2-2. Probability-weighted Annual Dose Rate from Individual Radionuclides Contributing to the Total Probability-weighted Annual Dose Rate from Igneous Disruption

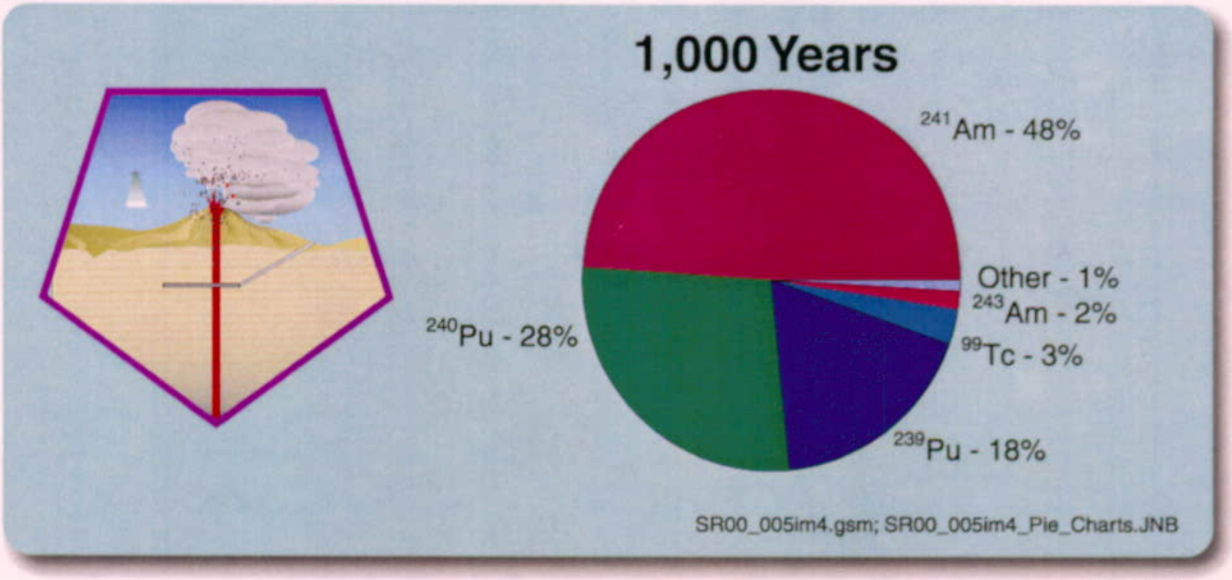

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Figure 4.2-3. Relative Contributions of Individual Radionuclides to the Total Probability-Weighted Igneous Dose Rate at 1,000 Years after Potential Repository Closure

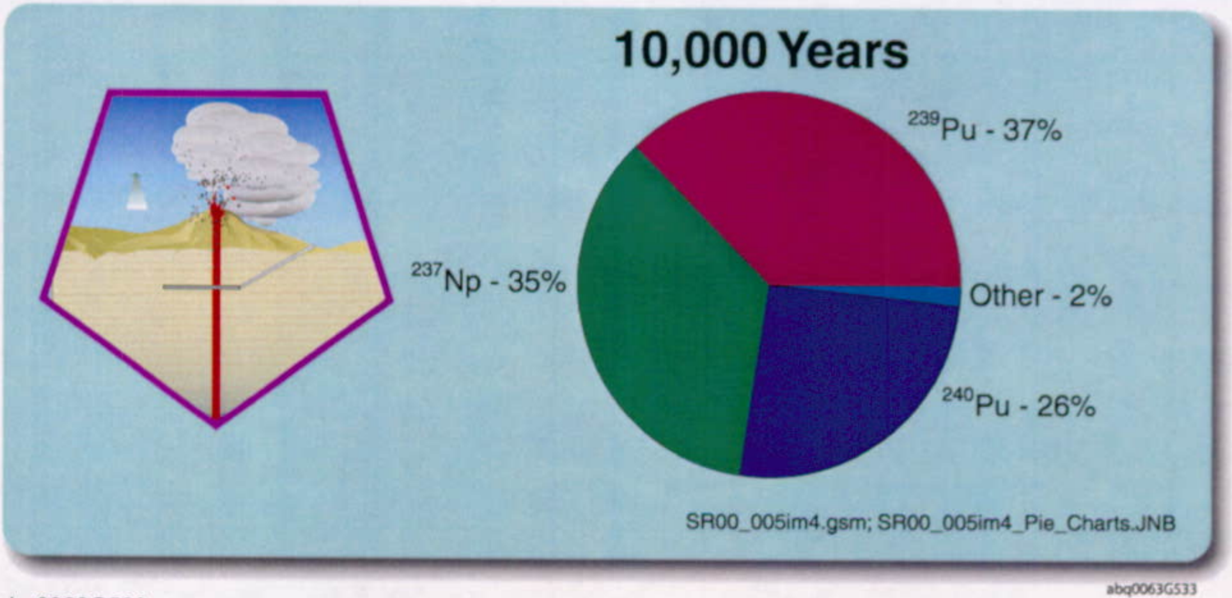

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Figure 4.2-4. Relative Contributions of Individual Radionuclides to the Total Probability-Weighted Igneous Dose Rate at 10,000 Years after Potential Repository Closure

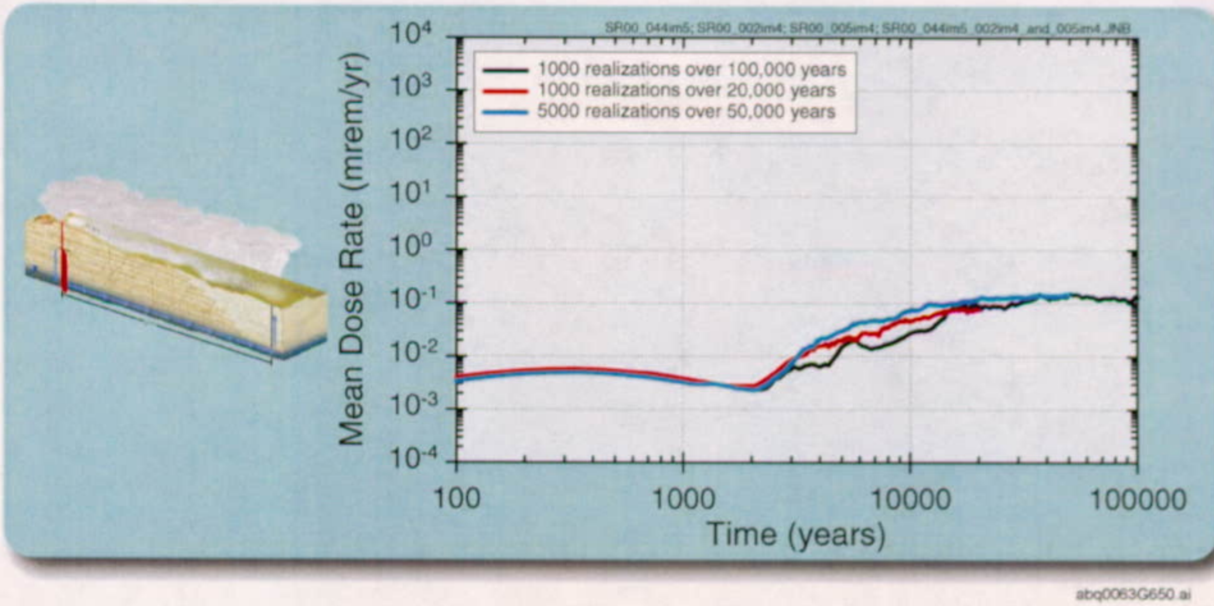

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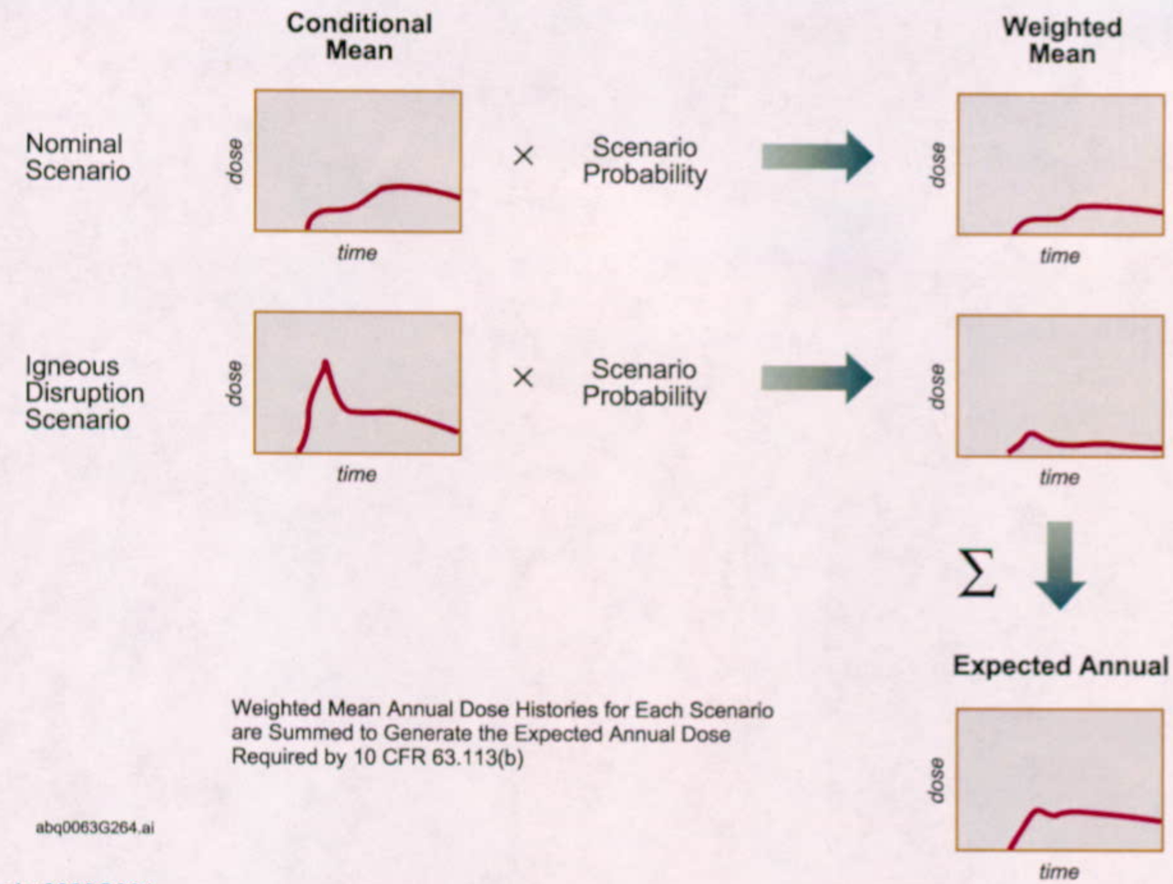
Figure 4.2-5. Comparison of Three Analyses Showing Sensitivity of the Probability-Weighted Mean Annual Dose Rate Following Igneous Disruption to the Number of Realizations and Duration of the Simulation



abq0063G264

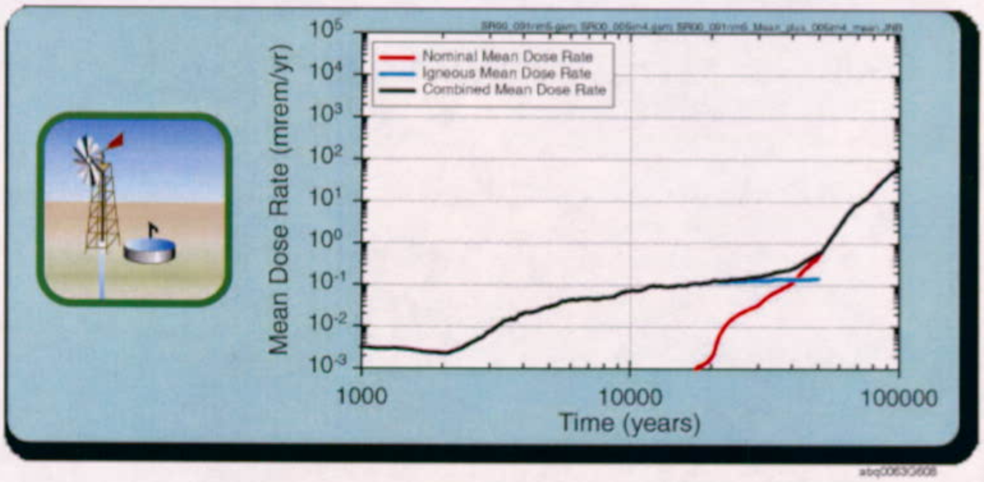
Source: Adapted from CRWMS M&O 2000 [147323], Figure 4.4-3

Figure 4.3-1. A Possible Method for Combining Scenarios

abq0063G456

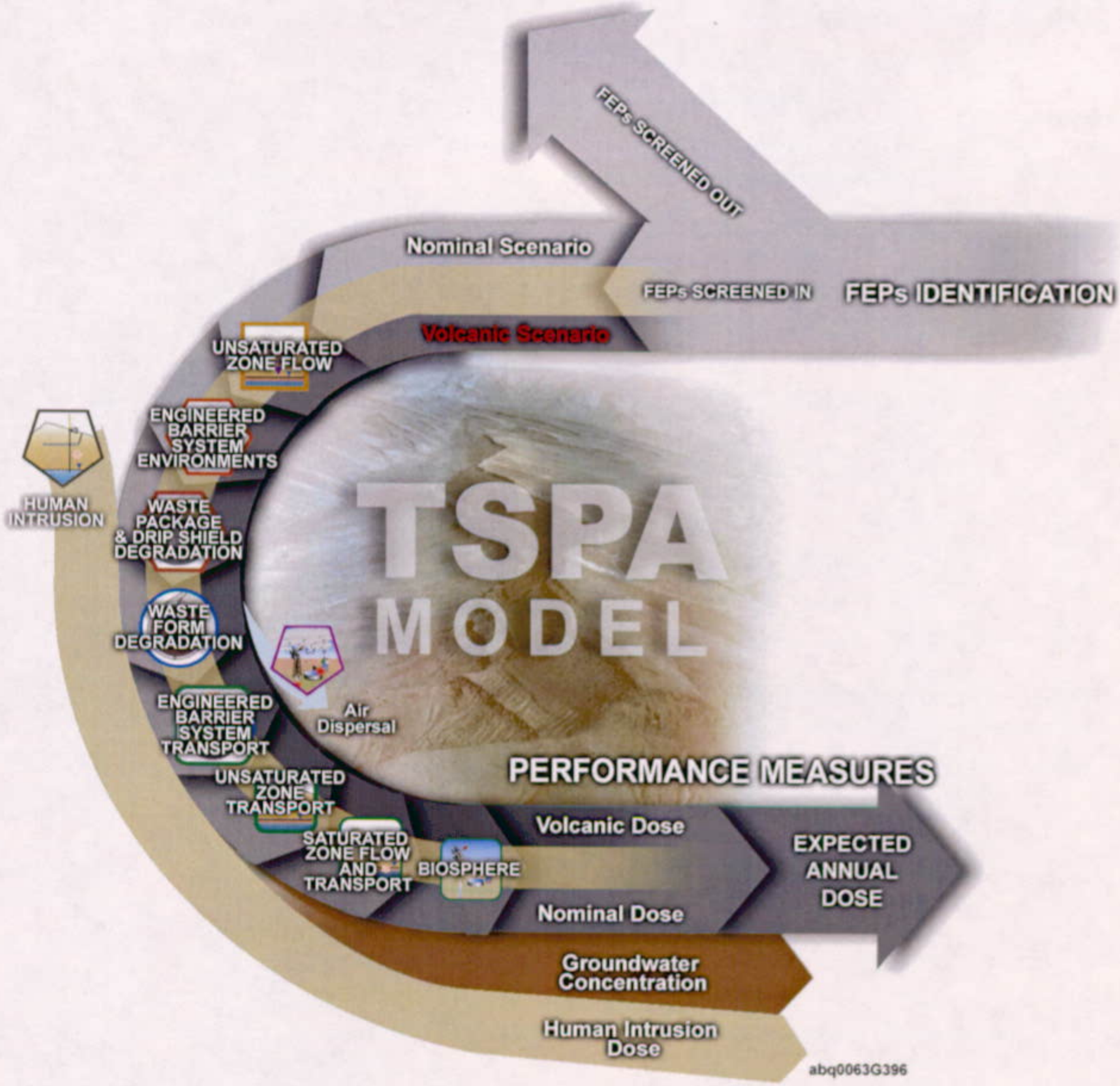
NOTE: Igneous Disruption Scenario Does Not Include Nominal Waste Package Failures

Figure 4.3-2. Method for Combining Scenarios Used in this Total System Performance Assessment



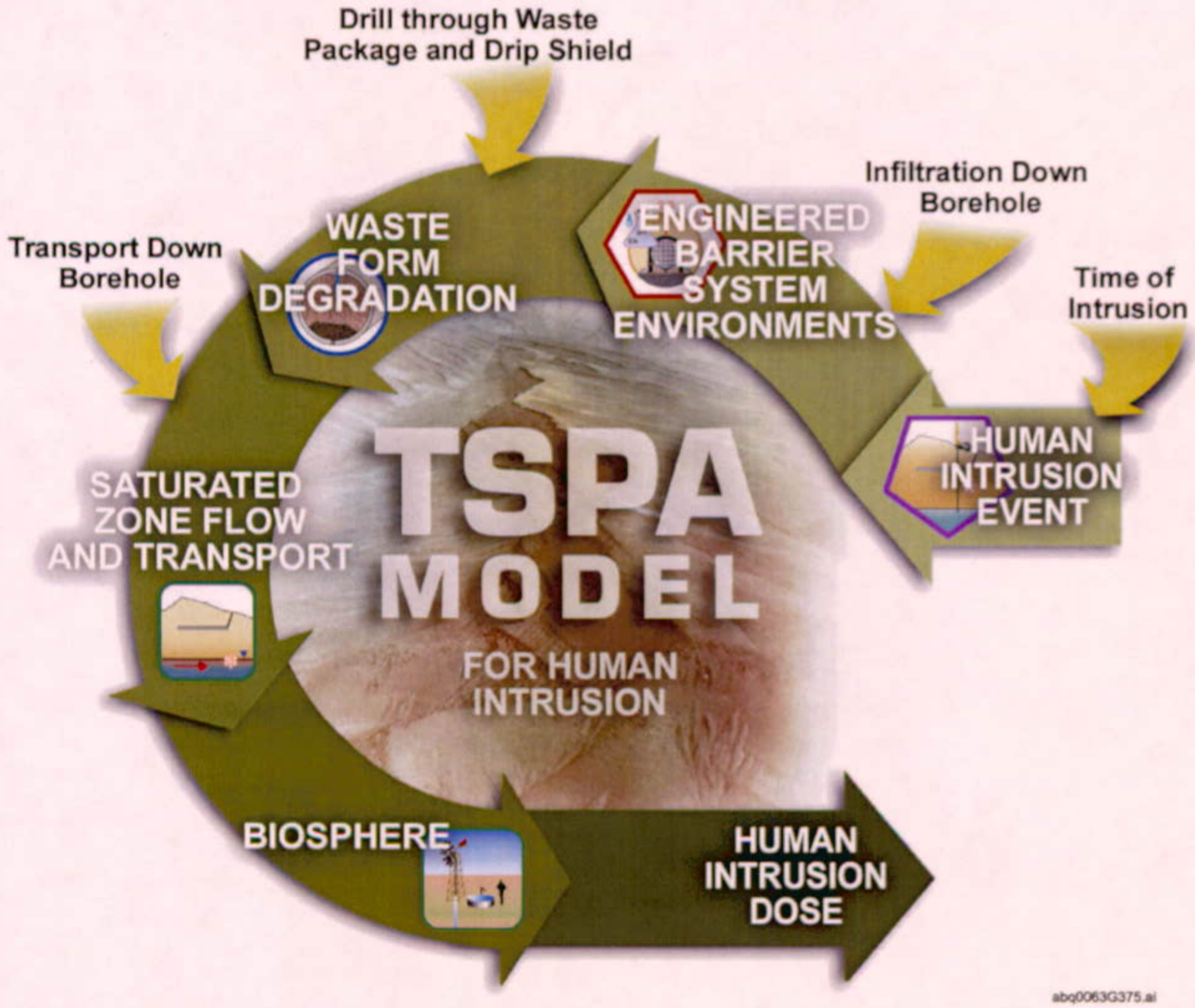
abq0063G608

Figure 4.3-3. Time Histories of Mean Annual Dose for Combined Nominal Plus Disruptive Results



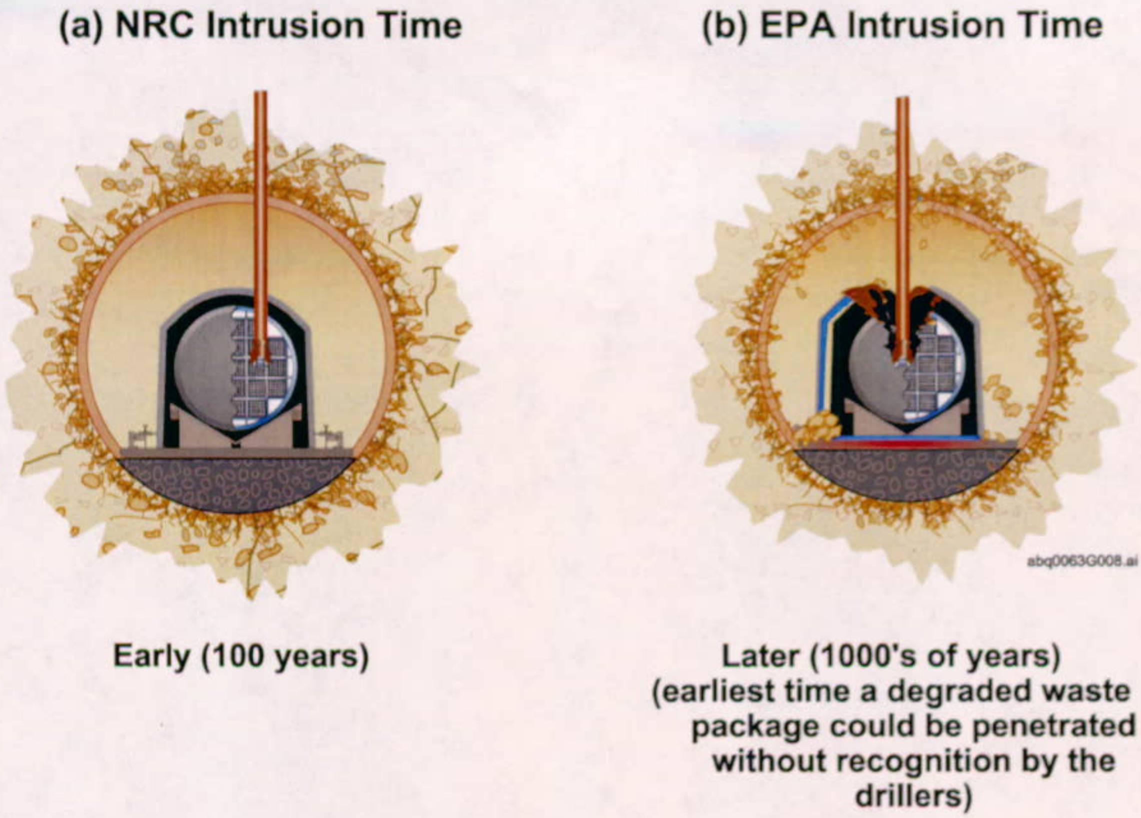
abq0063G396

Figure 4.4-1. Relationship of the Separate Human Intrusion Performance Assessment to Total System Performance Assessment-Site Recommendation



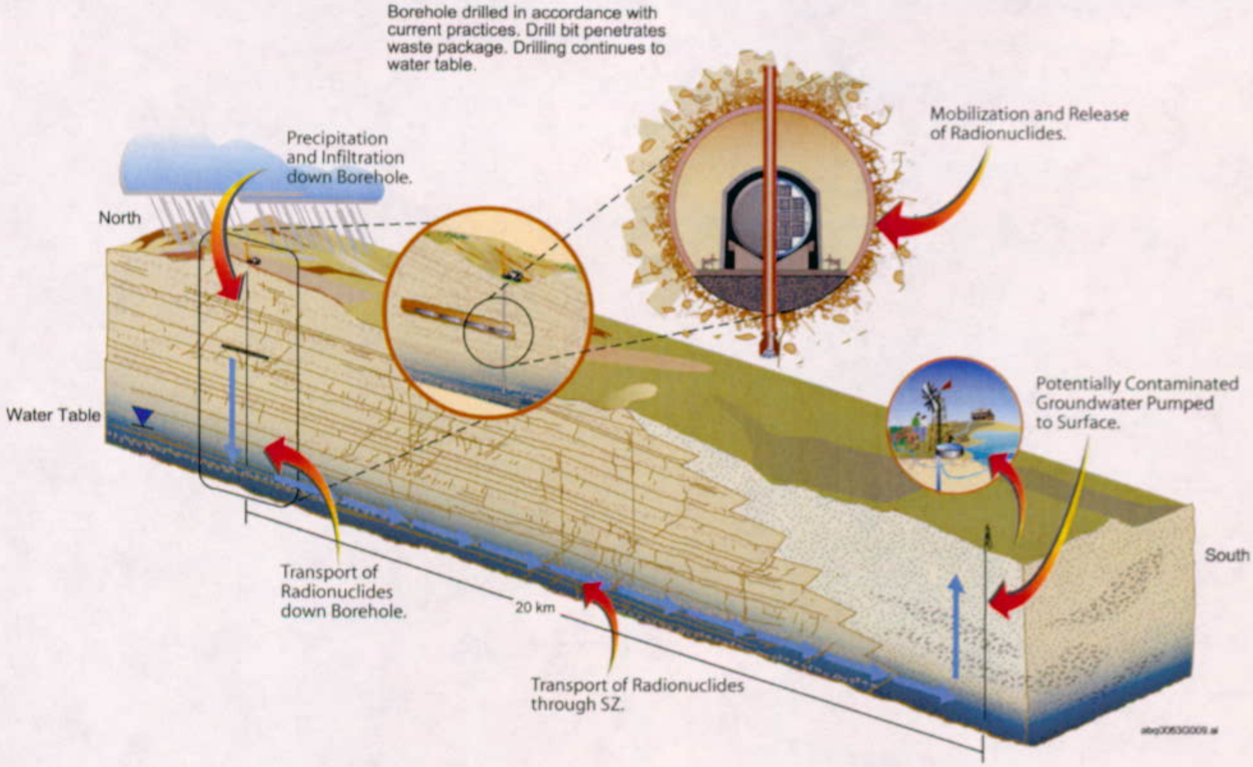
abq0063G375

Figure 4.4-2. The Human Intrusion Scenario in Total System Performance Assessment-Site Recommendation



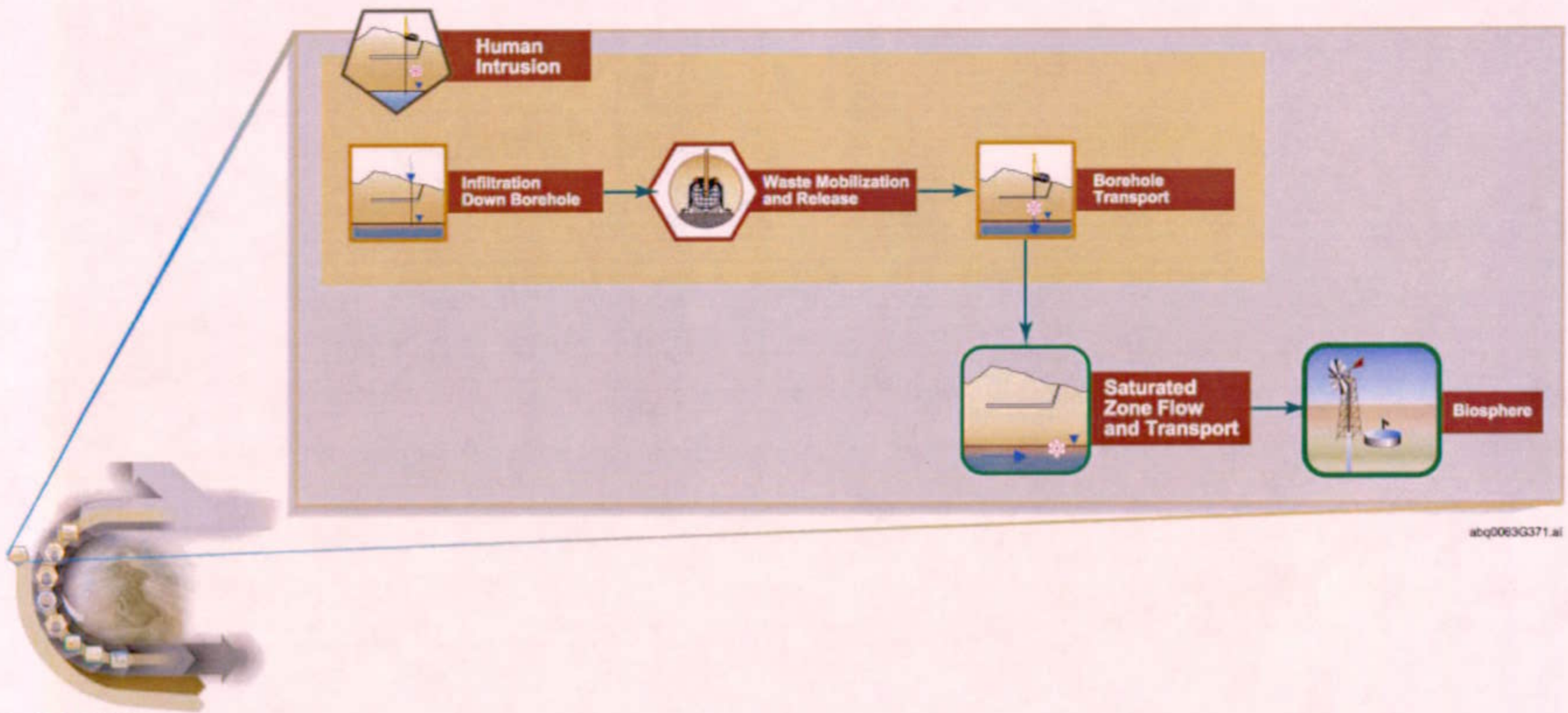
abq0063G008

Figure 4.4-3. Schematic Illustration of Intrusion Time as Specified in (a) NRC Regulations, and (b) EPA Regulations



abq0063G009

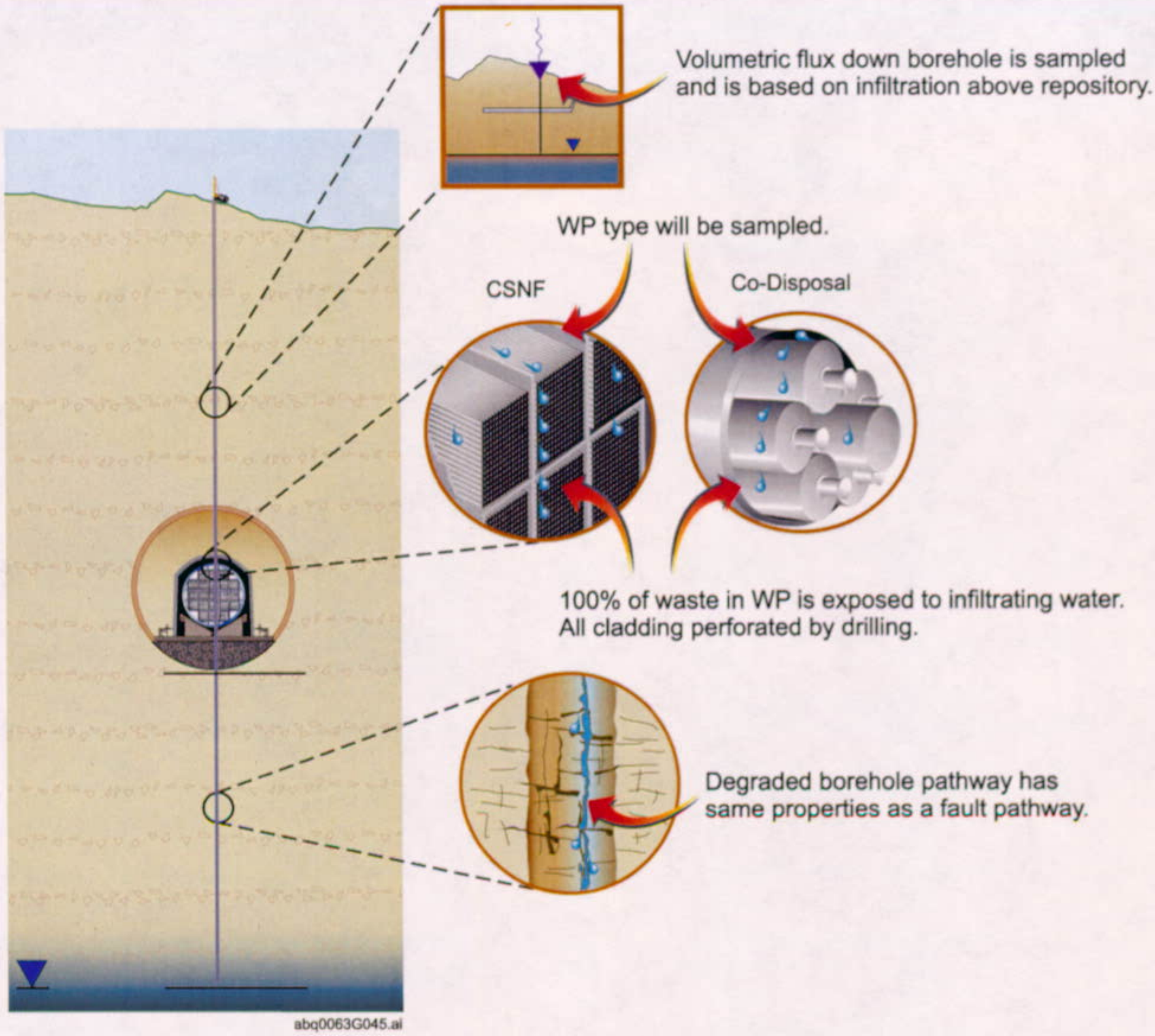
Figure 4.4-4. Conceptualization of Human Intrusion for Total System Performance Assessment-Site Recommendation



abq0063G371

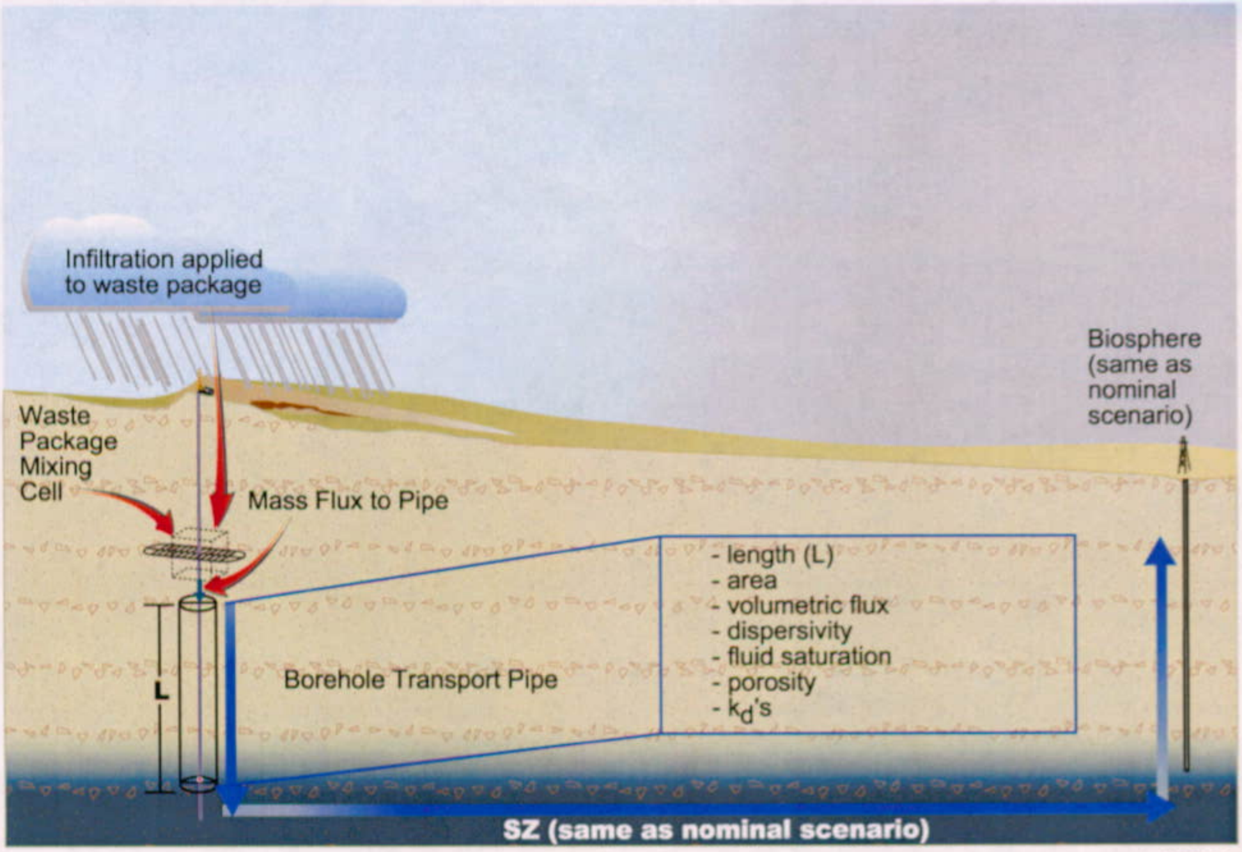
NOTE: Index figure in lower left is same as Figure 2.1-5

Figure 4.4-5. Key Components of Human Intrusion for Total System Performance Assessment-Site Recommendation



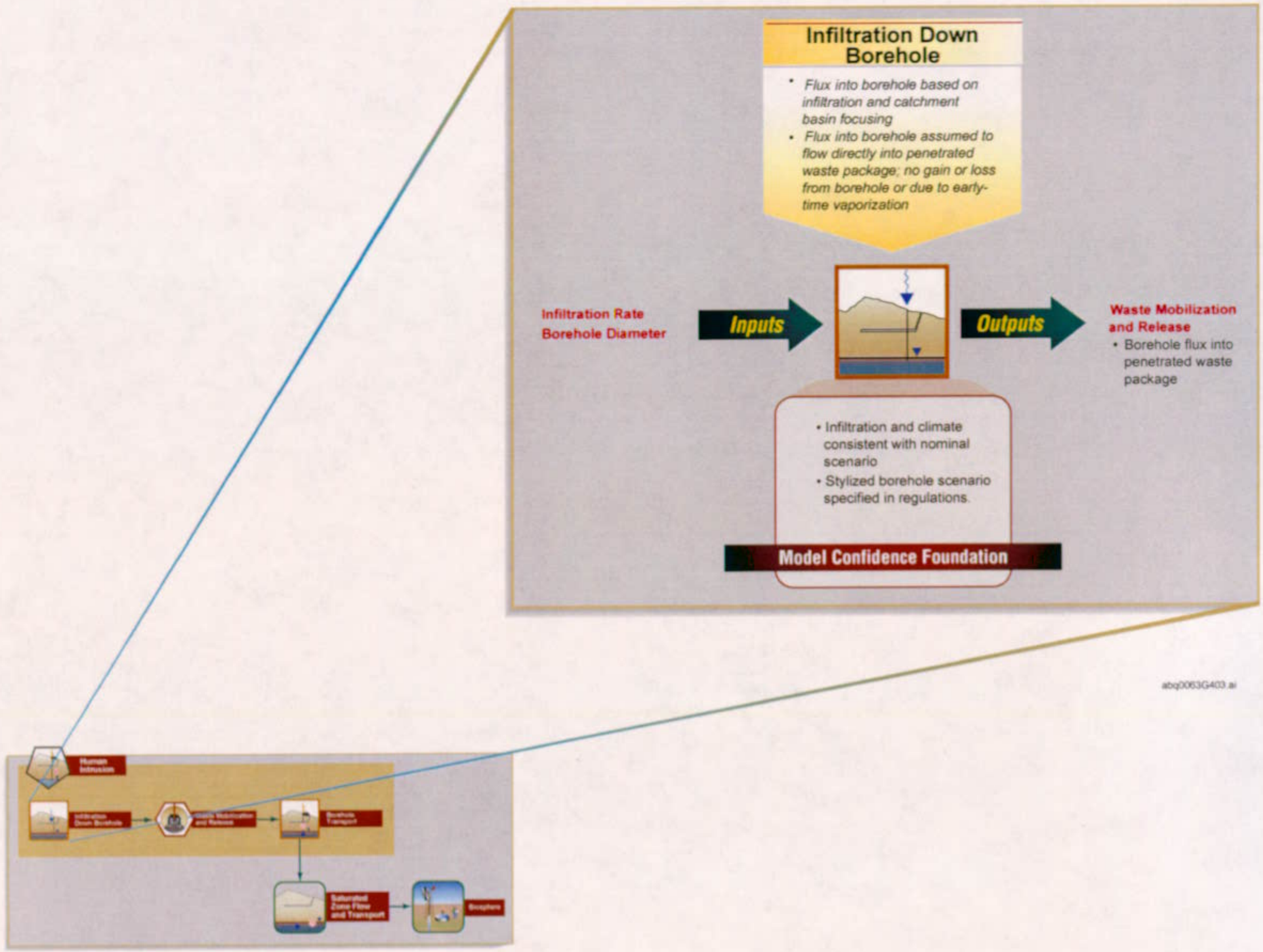
abq0063G045

Figure 4.4-6. Key Technical Assumptions for Human Intrusion in Total System Performance Assessment-Site Recommendation



abq0063G073

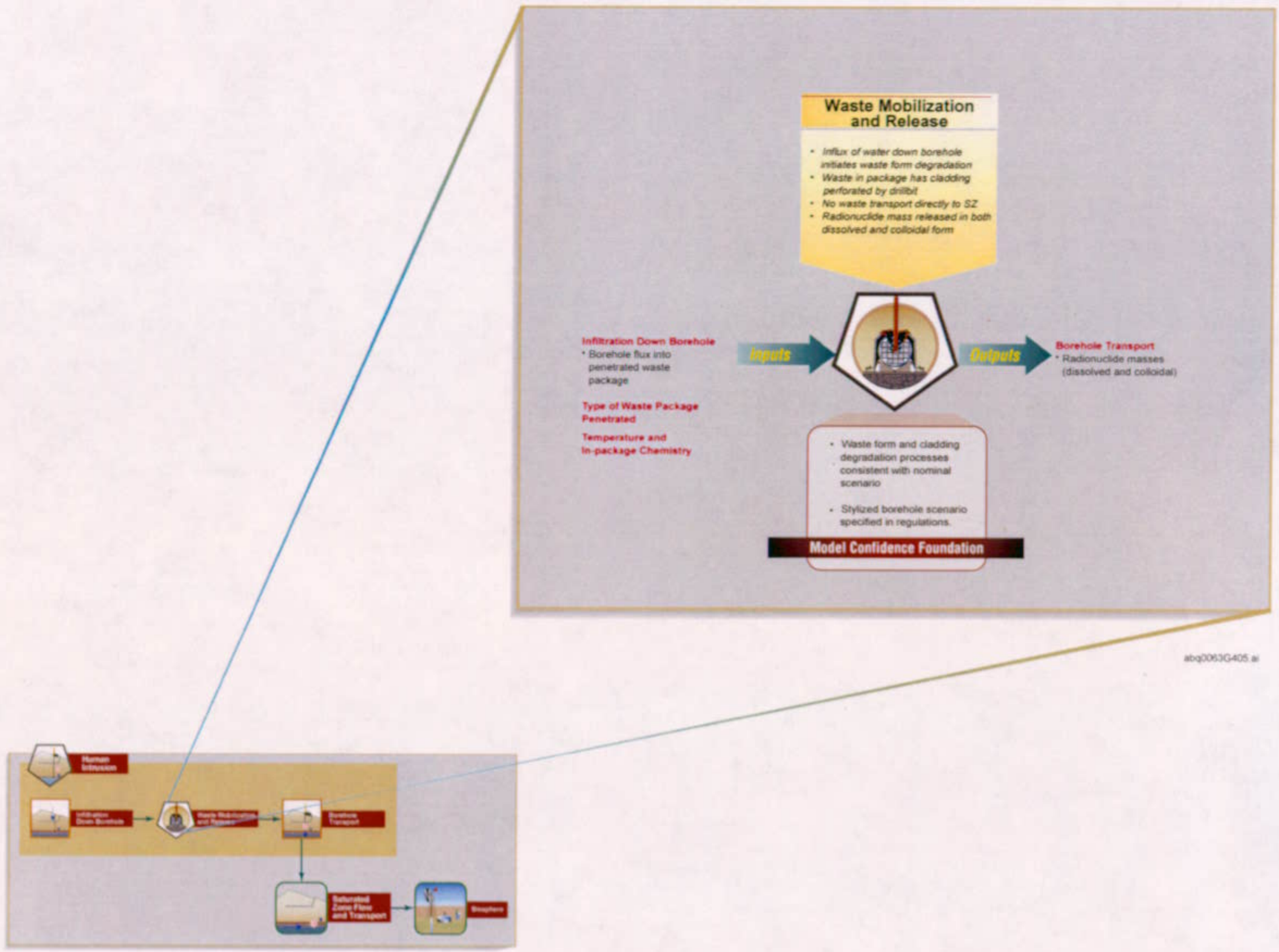
Figure 4.4-7. Implementation of Human Intrusion in Total System Performance Assessment-Site Recommendation



abq0063G403

NOTE: Index figure in lower left is same as Figure 4.4-5

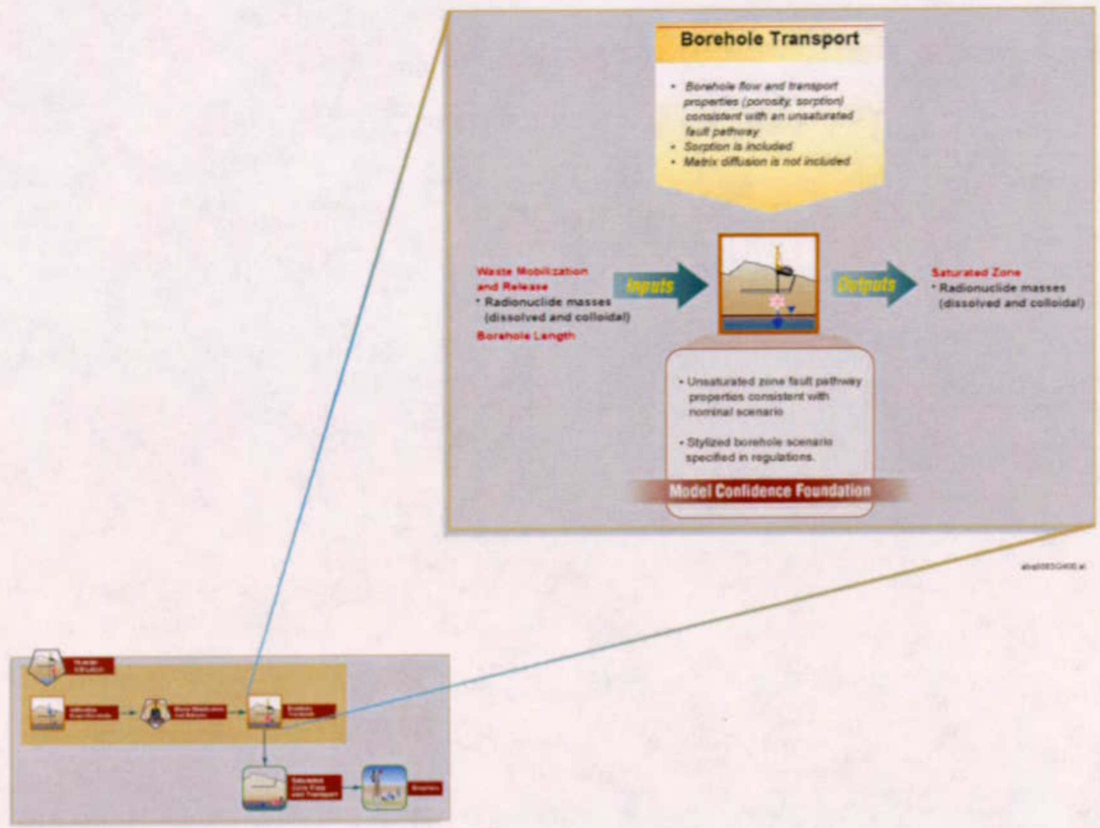
Figure 4.4-8. Coupling of the Infiltration down Borehole Component to other Human Intrusion Components



abq0063G405

NOTE: Index figure in lower left is same as Figure 4.4-5

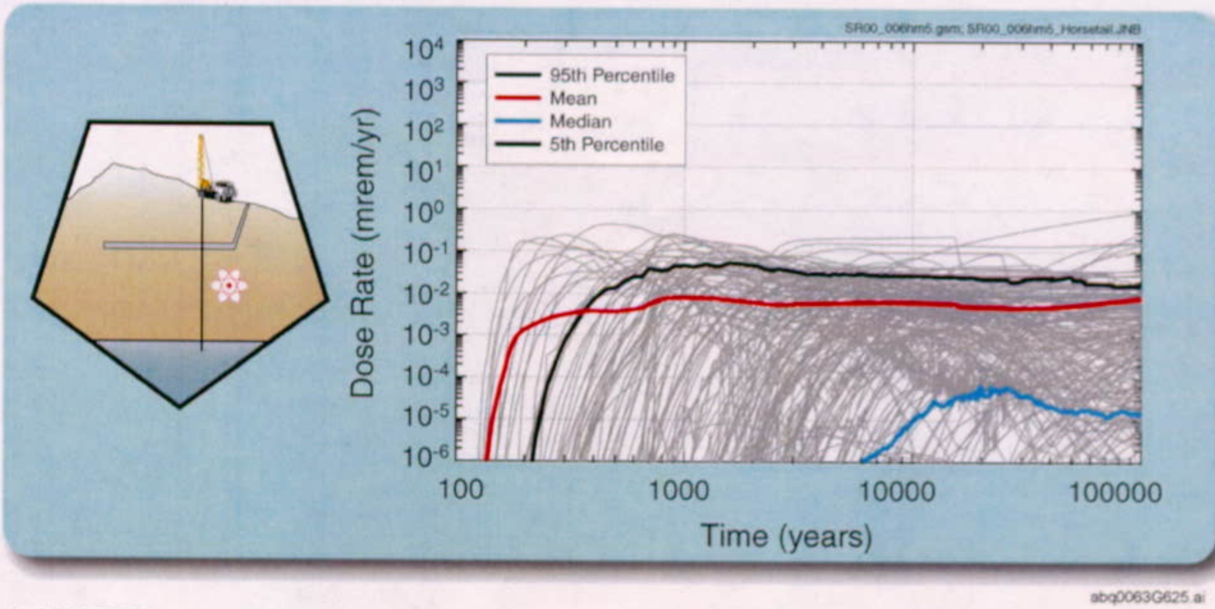
Figure 4.4-9. Coupling of the Waste Mobilization and Release Component to other Human Intrusion Components



abq0063G406

NOTE: Index figure in lower left is same as Figure 4.4-5

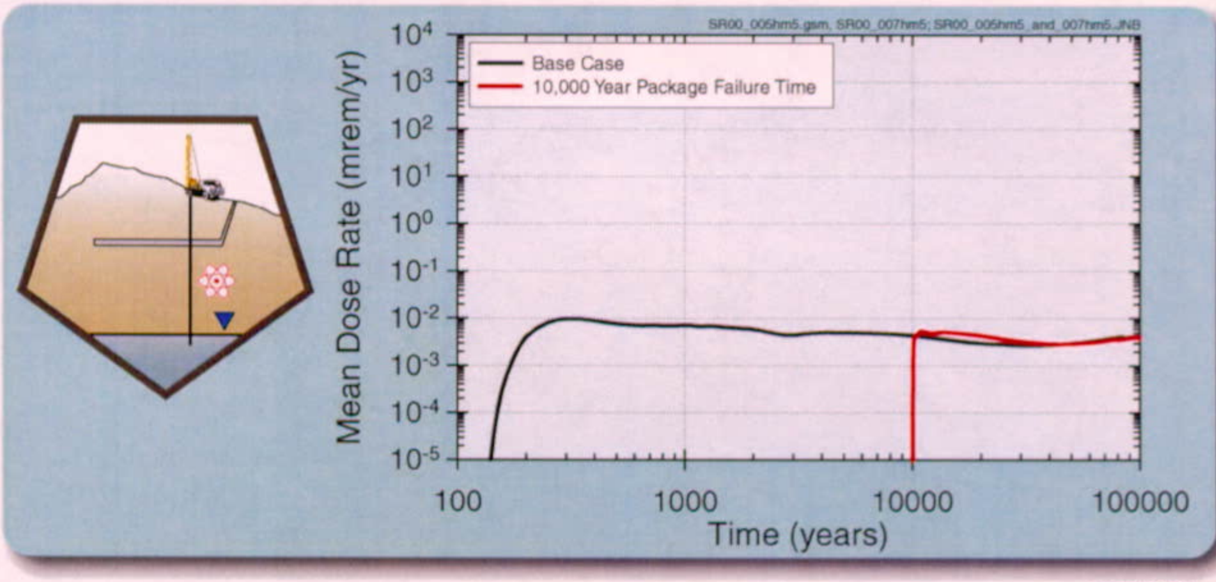
Figure 4.4-10. Coupling of the Borehole Transport Component to other Human Intrusion Components



abq0063G625

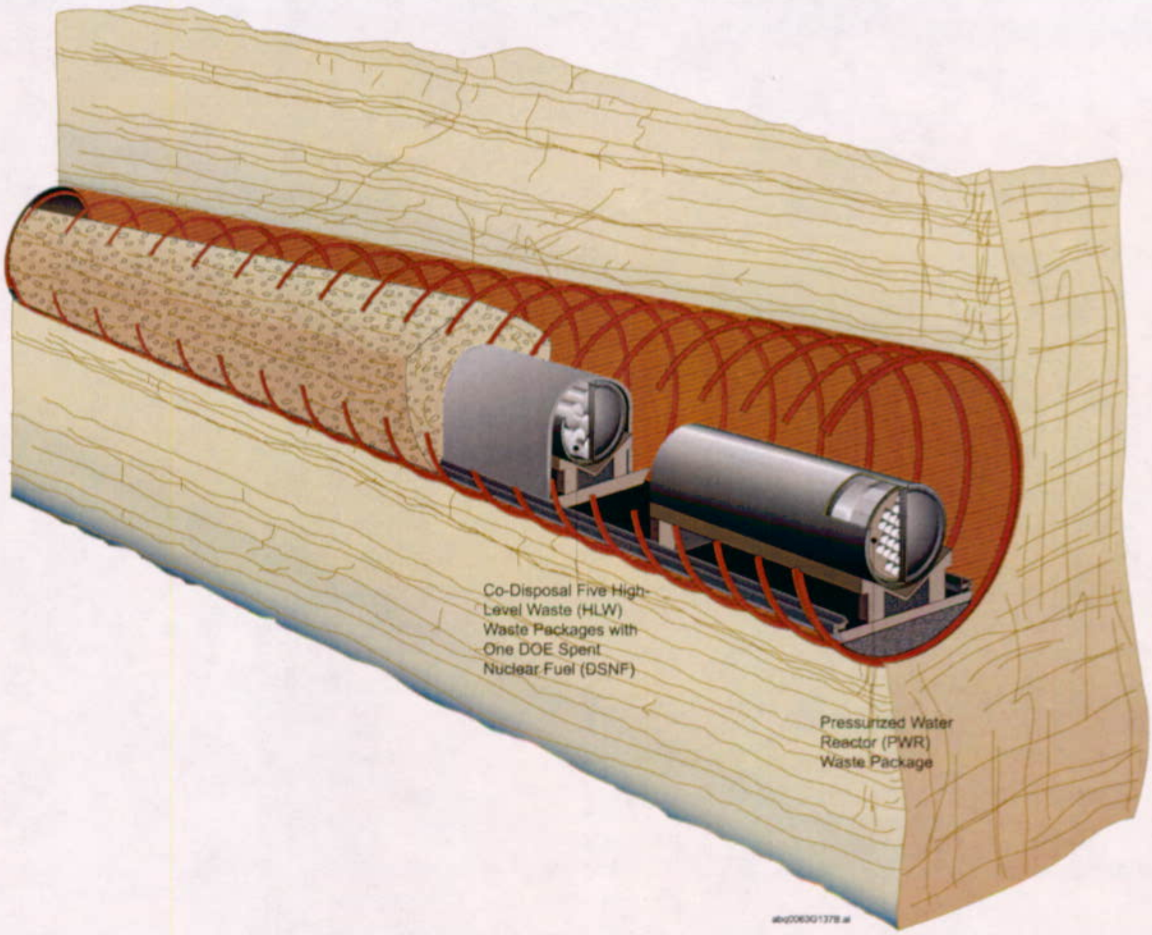
NOTE: The results from each of the 300 realizations are shown along with the mean, median, 95th, and 5th percentile curves

Figure 4.4-11. Simulated Dose Rate Histories for the Human Intrusion Scenario



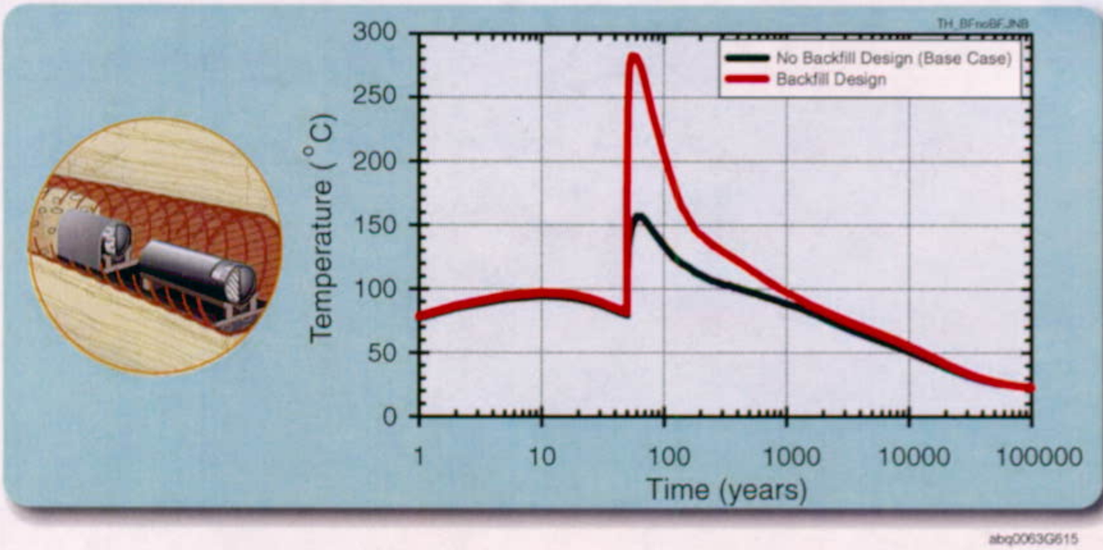
abq0063G618

Figure 4.4-12. Comparison of Total Mean Dose Histories for Human Intrusion Scenarios with Intrusions at 100 Years and 10,000 Years after Potential Repository Closure



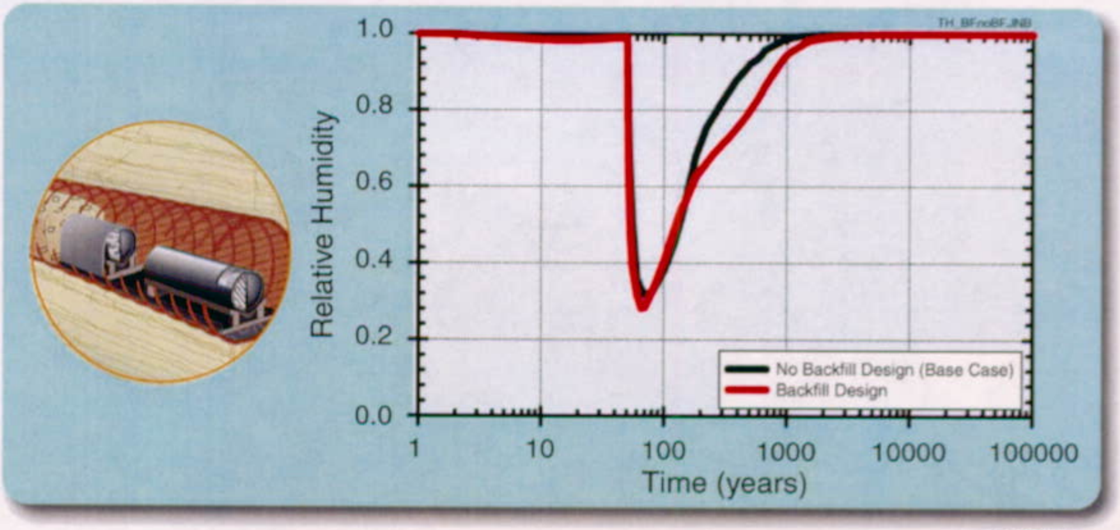
abq0063G137B

Figure 4.6-1. EBS Design with Backfill



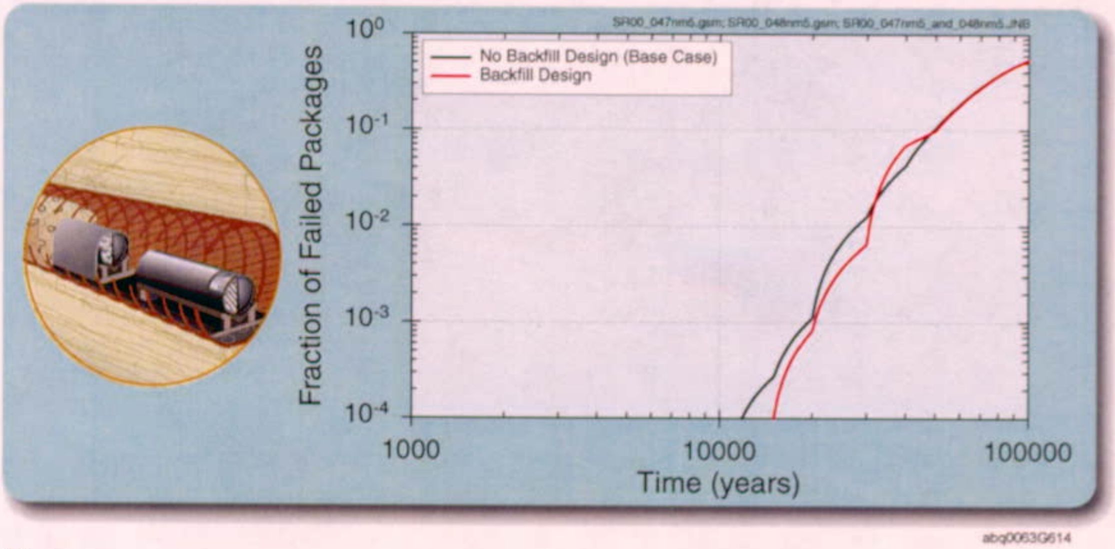
abq0063G615

Figure 4.6-2. Comparison of Waste Package Temperature for no Backfill and Backfill Cases (Nominal Scenario)



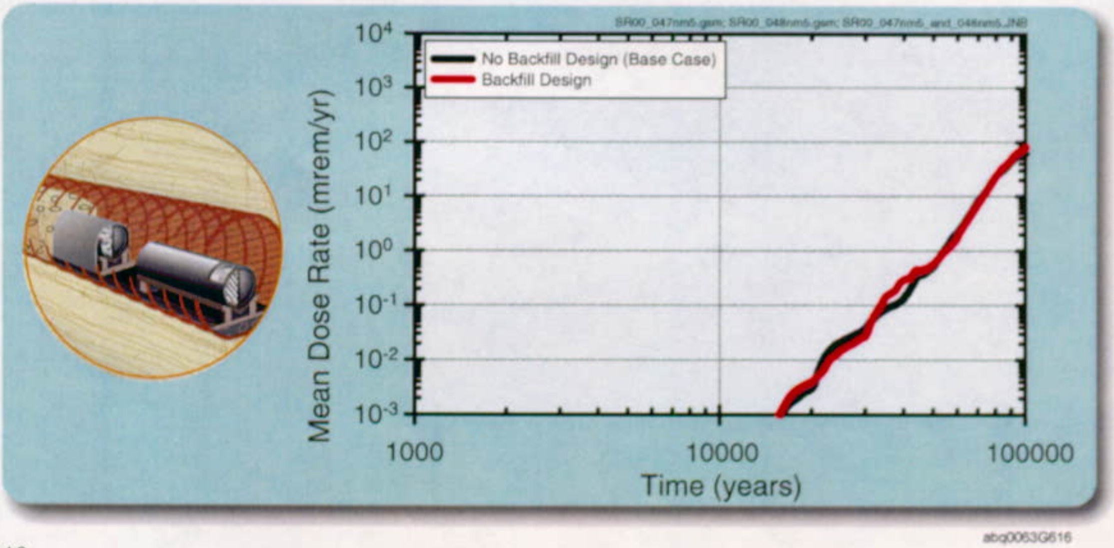
abq0063G613

Figure 4.6-3. Comparison of Relative Humidity for no Backfill and Backfill Cases (Nominal Scenario)



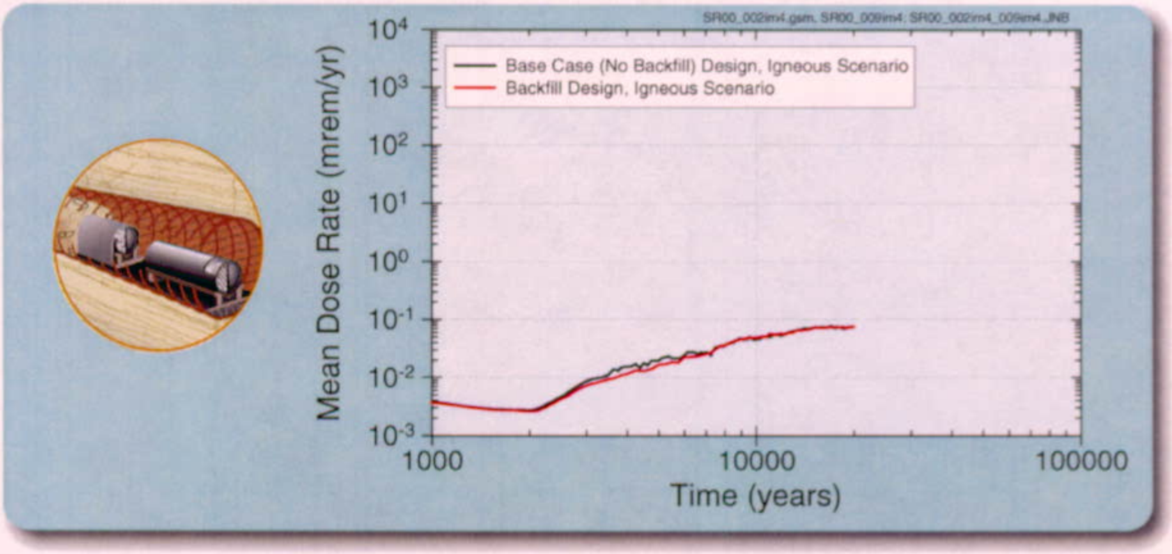
abq0063G614

Figure 4.6-4. Comparison of Initial Waste Package Failure for no Backfill and Backfill Cases (Nominal Scenario)



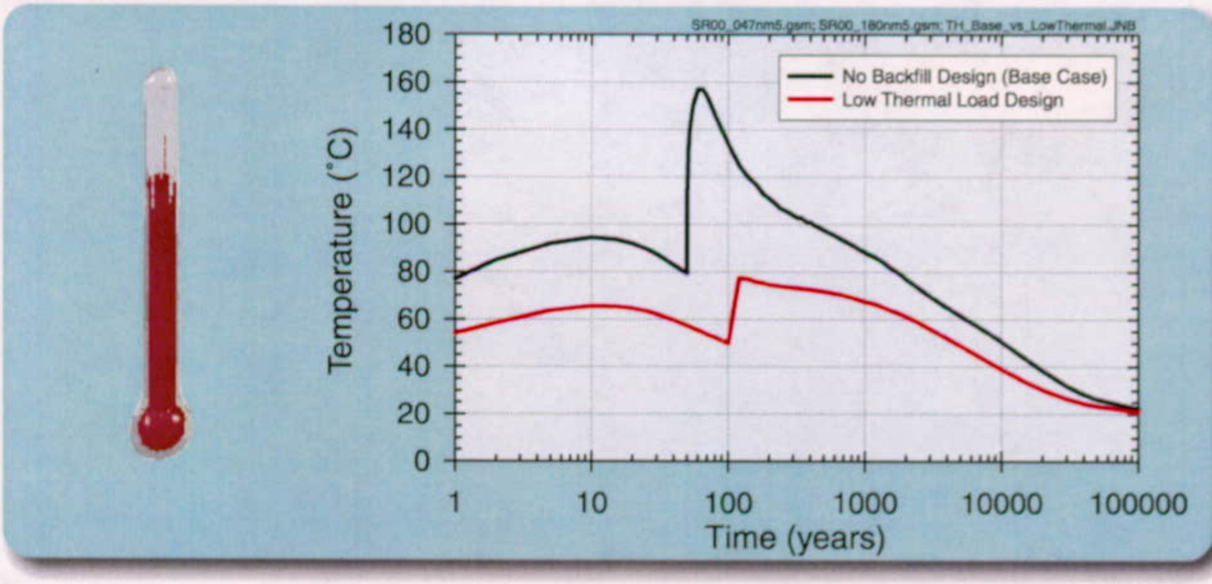
abq0063G616

Figure 4.6-5. Comparison of Dose for no Backfill and Backfill Cases (Nominal Scenario)



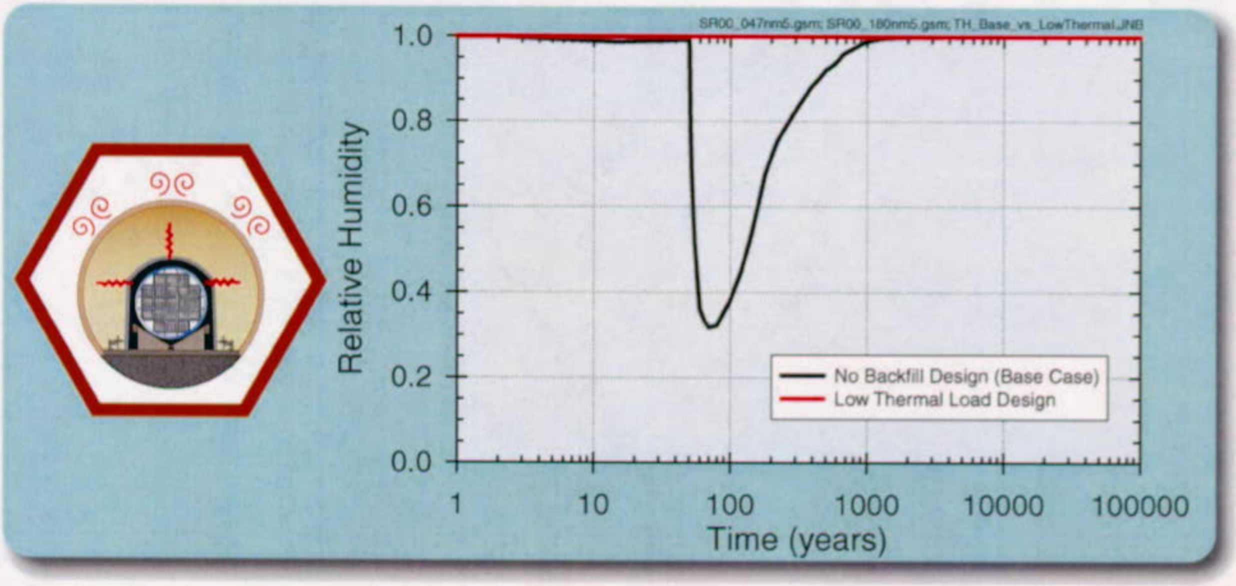
abq0063G617

Figure 4.6-6. Comparison of Dose for no Backfill and Backfill Cases (Igneous Scenario)



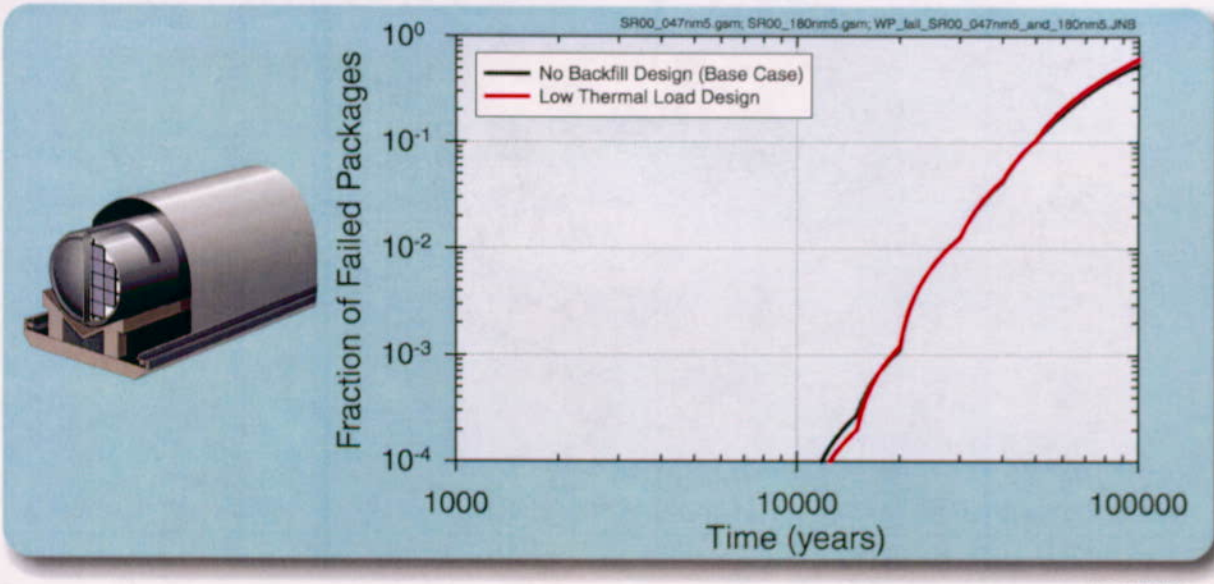
abq0063G646

Figure 4.6-7. Comparison of Waste Package Temperature for no Backfill and Low Thermal Load Design (Nominal Scenario)



abq0063G645

Figure 4.6-8. Comparison of Relative Humidity for no Backfill and Low Thermal Load Cases (Nominal Scenario)



abq0063G644

Figure 4.6-9. Comparison of Initial Waste Package Failure for no Backfill and Low Thermal Load Design Case (Nominal Scenario)

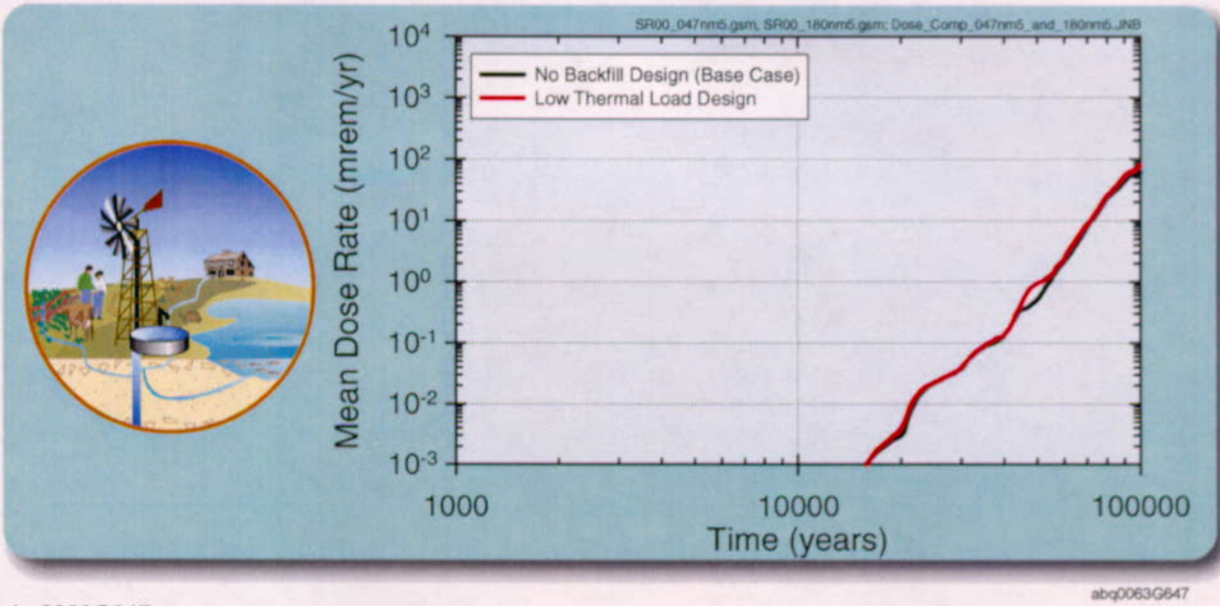


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Figure 4.6-10. Comparison of Dose for no Backfill and Low Thermal Load Design Cases (Nominal Scenario)

5. SENSITIVITY ANALYSES

The uncertainty in and sensitivity of the total system performance analyses of the postclosure period for the potential Yucca Mountain repository are presented in this section. The previous section provided a detailed evaluation of the nominal scenario, the igneous scenario, and the result of combining these scenarios, as well as the stylized human intrusion scenario and analyses of alternative designs. This section provides three fundamental sections that allow a greater look at the uncertainty in those results, as well as sensitivity analyses to evaluate which parameters or features of the potential repository system are most important in explaining the overall performance.

Section 5.1 provides an evaluation of the stochastic variables that are most significant in determining the output of the TSPA-SR model. This "uncertainty importance analysis" is obtained by utilizing various statistical methods to identify the most important contributors to the spread in the overall model results, and to identify contributors to the extreme, or outlier, outcomes in the model results. These analyses are presented for both total system and intermediate performance results for the nominal and igneous scenarios. As identified in the discussion on design impacts, the temperature is not a key contributor to the dominant failure mode for the waste package. Thus, temperature doesn't appear as a main contributor to potential repository performance here, and is not discussed in detail in subsequent sections 5.2 and 5.3.

Section 5.2 builds on the results of Section 5.1 and describes the results of additional TSPA-SR model simulations that are intended to demonstrate the effect of individual parameters or assumptions on the model results. For these analyses, various parameters were assigned fixed values (rather than allowed to vary within the assigned uncertainty range), or alternative modeling assumptions were made, and multiple realizations were conducted of these modified inputs. These analyses are often called "one-off" sensitivity analyses, due to the nature of changing a single parameter at a time. These analyses were conducted to evaluate sensitivity of many of the components of the nominal and igneous scenario models.

An additional type of sensitivity analysis is presented in Section 5.3. These robustness analyses were intended to help evaluate the performance of the potential repository system when the various natural and engineered barriers (to release and transport of radionuclides) are assumed to be degraded, either individually or in combination. These analyses were only conducted using the nominal scenario as the basis. These analyses were conducted for the many barriers within the geologic repository system. The degradation characteristics did not increase or decrease the range of uncertainty in the nominal case. It just evaluated the extremes of the parameter or model range already in the TSPA-SR model. These analyses were based on modifying barriers, not necessarily on evaluating parameters or models identified in Section 5.1, though the analyses often confirmed the results of Section 5.1.

5.1 UNCERTAINTY IMPORTANCE ANALYSIS

This section presents the results of analyses intended to identify those stochastic variables that have the greatest impact on the output of the TSPA-SR model. The goal is twofold: (a) use regression analysis to determine the most important contributors to the spread in probabilistic model results, and (b) use classification tree analysis to identify those variables controlling extreme outcomes in the full suite of probabilistic results. Details of these techniques, presented in Section 2.2.5, are summarized below for the reader's benefit.

Regression analysis–is a tool for quantifying the strength of input-output relationships in the TSPA-SR model. To this end, a stepwise linear rank regression model is fitted between total dose at a given time (or some other performance measure) and all randomly sampled input variables. Parameters are ranked on the basis of how their exclusion would degrade the explanatory power of the regression model. The importance ranking metric used for this purpose is the uncertainty importance factor, which is defined as the loss in explanatory power (R^2-loss) divided by the coefficient of determination (R^2) of the regression model. Note that the uncertainty importance factor also quantifies what proportion of the total spread (variance) in total dose explained by the regression model can be attributed to the variable of interest.

Classification tree analysis–a subset of classification and regression tree analysis, is a method for determining what variables or interactions of variables drive output into particular categories. Classification and regression tree analysis can be used to generate decision rules that determine whether a particular realization would produce "high" or "low" dose depending on the values of the most important variables. Unlike regression analysis, which is based on the total range of model outcomes, classification tree analysis focuses on extreme values of model results and tries to relate them to specific ranges of values for the important variables.

Each TSPA-SR model run, as discussed in Section 4.1, is typically based on a 300-realization sample. Thus, for any given time, there are 300 output values (e.g., dose) as well as 300 input values for each of the stochastic input variables. For the purposes of uncertainty importance analysis, these results are post-processed in three different ways. The most common approach is to consider the spread in dose at a fixed time slice, and use regression analysis to identify the primary contributors to this spread in dose. A second approach is to consider the spread in time to reach a specified dose level, and determine the key contributors to this spread also using regression analysis. The third approach is to partition the results at any given time into "high" or "low" dose categories, and then use classification tree analysis to identify the variables influencing this separation into categories. The results presented in this section will use all three of the approaches discussed above.

The results that follow are of uncertainty importance analysis for the nominal scenario results with total dose as the performance measure. This is followed by an analysis of various intermediate performance measures (e.g., EBS and SZ release for individual radionuclides, percent waste packages failing, etc.) for the nominal scenario. Finally, the igneous scenario results are analyzed with total dose as the performance measure. Note that unless otherwise indicated, these analyses reflect projected performance of the disposal system over 100,000 years.

5.1.1 Nominal Scenario, Total Dose

Section 4.1 presents a discussion of the range of dose that the average member of the critical group is likely to receive over 100,000 years for the nominal scenario. Figure 4.1-5 shows that there is a broad range in projected dose rates at any given time, with most of the realizations producing negligible dose during the first few tens of thousands of years. Regression analysis of such data is difficult because of the limited number of non-zero dose producing realizations, as well as the large number of stochastic inputs in the model. Therefore, regression analysis was restricted to those time slices (at 10,000 year increments) for which at least 100 realizations produced dose in excess of 10^{-5} mrem/yr. For the nominal scenario, as shown in Figure 4.1-5, the first such time slice was 40,000 years. The choice of 10^{-5} mrem/yr was based on a visual examination of dose histograms at selected time slices to identify a consistent separation threshold between non-zero dose values and infinitesimal dose values. Requiring a minimum of 100 observations for carrying out the regression analysis was felt to be a reasonable choice for eliminating bias in building input-output models containing approximately 10 percent of the more than 240 stochastic inputs in the TSPA-SR model.

Regression analysis was carried out at 10,000-year increments, from 40,000 to 100,000 years. Stepwise rank regression models between total dose and a set of statistically significant input variables were built for each case, and the most important variables were identified based on the value of their uncertainty importance factor. Figure 5.1-1 shows a bar chart of these results generated for the 40,000-year data, indicating that the most important variables are the SCC outer and middle lid stress profiles and the Alloy-22 outer lid median general corrosion rate. Similar importance analyses results for 70,000 and 100,000 years are shown in Figure 5.1-2 and Figure 5.1-3, respectively. These also indicate that the waste package degradation related parameters such as the stress profiles and the median corrosion rates continue to be important over the longer time period. The natural system is beginning to affect the system response after about 70,000 years, albeit marginally, as indicated by the inclusion of SZ groundwater flux in the list of important variables.

Figure 5.1-4 shows how uncertainty importance factors change with time for the key uncertain variables. Over the first 100,000 years, the SCC outer lid stress profile is seen to be consistently the most important, followed by the median general corrosion rate of the Alloy-22 outer and middle lids. Note also the slow but steady increase in importance for SZ groundwater flux with time. Figure 5.1-5 presents a different interpretation of the same information in terms of time-averaged composite uncertainty importance ranking. The importance ranking (with the most important variable getting the highest rank) from all time slices is averaged and normalized such that a (perfect) rank of 1 indicates that the corresponding variable (e.g., SCC outer lid stress profile) is consistently the most important at all time slices.

As discussed in Section 2.5, another tool for investigating the relationship between model output and key uncertain inputs is the use of scatter plots. Figure 5.1-6 shows scatter plots at 40,000, 70,000, and 100,000 years between total dose and SCC outer lid stress profile, and total dose and Alloy-22 outer lid median general corrosion rate. In general, an increase in the value of these variables leads to a greater dose, although there is considerable scatter in the data. This suggests that no single uncertain input is overly dominant in affecting the spread of total dose in the current TSPA-SR model.

Next, we present an application of classification tree analysis to the same data sets to determine which variables control extreme outcomes. The 300 realizations of total dose, at any given time slice, are first categorized as "high" if the values are in the top 10 percentile, or "low" if the values are in the bottom 10 percentile. The classification tree analysis algorithm then determines which variables are most capable of explaining the separation of these realizations into the appropriate categories.

Figure 5.1-7 (top) shows a decision tree summarizing the classification tree analysis in terms of the two most important variables for the 40,000-year data. In this case, much of the separation into high and low dose values can be explained on the basis of a single variable, the SCC middle lid stress profile. The second most important variable, uncertainty in agricultural water usage per farm, provides only marginal additional explanatory power, and that too, for the low dose values. Figure 5.1-7 (bottom) is a partition plot of the same data, showing where the high and low dose producing clusters actually occur in the bivariate parameter space of the two most important variables. Note that the partition plot is actually an input-input scatter plot, with the data points color coded to indicate which category they belong to, and the dividing lines indicating where the split between the categories occurs.

Note that these variables are important only in terms of explaining the decision rules (e.g., *if* variable_1 > x *and* variable_2 < y *then* dose high) leading to extreme outcomes. The variable importance ranking obtained from regression analysis provides a complementary piece of information, namely, which variables contribute the most to the overall spread in outcomes. Thus, agreement between the two sets of importance rankings should not be expected to be perfect for all cases. The partition plot essentially supplements the insight drawn from the regression analysis in that one can determine how exactly an important uncertain input is affecting the dose.

Classification tree analysis results for the 70,000-year dose data are summarized in the decision tree shown in Figure 5.1-8 (top). Here, the variables related to SCC middle and outer lid stress profiles provide the most explanatory power in the categorization problem. High values for both the stress profiles leads to high doses, and conversely, low values for both stress profiles leads to low doses. This trend is also demonstrated in the partition plot shown in Figure 5.1-8 (bottom), where high and low dose producing outcomes are separated into the top right and bottom left quadrants.

In Figure 5.1-9 (top), the categorization decision tree for the 100,000 year data is shown. As in the case of the 40,000 year data, a single variable (SCC outer lid stress profile) is adequate for explaining most of the separation between the two categories. Figure 5.1-9 (bottom) confirms the dominating influence of this variable using the partition plot.

Thus far, the probabilistic results have been analyzed in terms of the spread in total dose at any given time. A second way of examining the same data is to analyze the results in terms of the spread in the time to reach a given dose. Thus, instead of "slicing" the multi-realization dose versus time data vertically along the time axis, the data are sliced horizontally along the dose axis. The dependent variable for the regression model becomes the time to reach a given dose, while the independent variables remain the same set of stochastic inputs used earlier.

This analysis was carried out at 4 specified dose levels, 0.01 mrem/yr, 0.1 mrem/yr, 1 mrem/yr, and 10 mrem/yr. Stepwise regression was carried out using time (rather than dose) as the dependent variable, and uncertainty importance factors were calculated as before. The results indicate that the same set of waste package degradation-related parameters identified earlier dominate the uncertainty in the output. Therefore, for reasons of brevity, Figure 5.1-10 presents only the graphs showing how uncertainty importance factor changes with the specified dose level for the key uncertain variables. Comparing Figure 5.1-10 with Figure 5.1-4 does confirm the general similarity in importance ranking irrespective of whether the contribution to spread in dose or spread in time is used as the ranking metric.

An analysis of uncertainty importance based on a 1-million-year simulation is presented (Figure 4.1-19). Dose values from each of the 300 realizations were regressed against the set of stochastic inputs at 100,000-year increments. The results, shown in Figure 5.1-11, depict the decreasing importance of the EBS parameters, and the increasing importance of the natural system, as compared to the importance rankings shown previously for the first 100,000 years in Figure 5.1-4. Note, in particular, how infiltration scenario becomes the single-most dominant variable at around 200,000 years and continues to remain so over the 1-million-year simulation period. It should be pointed out that the quantitative importance rankings for variables other than infiltration scenario are not very reliable, especially in the 400,000 to 800,000 year time frame, because the regression analyses provided relatively poor fits to these data sets.

5.1.2 Nominal Scenario, Intermediate Results

This section focuses on the projected spread in intermediate results such as the waste package failure distribution and the mass release at various "pinch points." A pinch point is a location at which mass (or energy) is being transferred from one modeling domain (or subsystem or barrier) to another. ^{237}Np and ^{99}Tc were selected for this purpose because of their widely differing sorption characteristics and also because these two radionuclides are major contributors to the total dose (Figure 4.1-6). The two pinch points chosen for tracking mass release were the boundary between the EBS and the UZ, and the boundary between the SZ and the biosphere.

Using the standard regression analysis methodology with ^{99}Tc mass release as the dependent variable, we calculated uncertainty importance factors at those time slices for which at least 100 realizations yielded a mass release greater than 10^{-5} g/yr. The uncertainty importance factor time history for the 5 most important variables are shown in Figure 5.1-12 and Figure 5.1-13 for the EBS release and the SZ release, respectively. These figures show that the SCC outer and middle lid stress profiles, and the Alloy-22 outer and middle lid median general corrosion rates, continue to be important as in the case of total dose. Note, however, that a new variable, CSNF dissolution rate uncertainty, becomes the most important variable at 100,000 years for both pinch points (UZ and SZ).

Corresponding results with ^{237}Np EBS mass release as the dependent variable are shown in Figure 5.1-14. Once again, waste package degradation related parameters dominate in the importance ranking. However, once the radionuclide enters the natural system, its sorption characteristics have a significant effect. This is reflected in the importance ranking for the mass release from the SZ (Figure 5.1-15), where neptunium K_d in the alluvium becomes one of the important variables, along with infiltration scenario and the SZ groundwater flux. Collectively, these variables control the amount of neptunium being retained within the natural system, which explains their importance with respect to the release from the SZ.

Next, an examination of the waste package failure distribution (Figure 4.1-9) can provide further insights into the workings of the TSPA-SR model. To this end, a scatter plot of total dose and fraction waste packages failed at 100,000 years is shown in Figure 5.1-16. In particular, there is a focus on those realizations where more than 80 percent of the packages have failed. The two shaded regions in the figure demarcate "high" dose outcomes (dose greater than 100 mrem/yr) from "low" dose outcomes (dose less than 15 mrem/yr). A classification and regression tree analysis of this data, as depicted in Figure 5.1-17 (top), indicates that infiltration scenario and SZ groundwater flux are the two most important parameters in explaining the categorization. The corresponding partition plot is shown in Figure 5.1-17 (bottom). The important point to note here is that even when a large percentage of the waste packages have failed, certain combinations of natural system parameters can yield dose below the 10,000-year regulatory limit even at 100,000 years. This analysis demonstrates the power of classification and regression tree analysis in "mining" the data to provide insights into cause-effect relationships that are not readily apparent from the results presented in Section 4.1, or the regression analyses presented in Section 5.1.1.

Returning to the issue of uncertainty importance as identified from regression analysis, it is useful to ask what parameters are driving system performance in addition to the waste package degradation related parameters already determined as critical uncertainties. For this purpose, a modification of the base case simulation was performed where the two SCC stress profile parameters and the two Alloy-22 median general corrosion rates (i.e., one for the middle lid and one for the outer lid) were fixed at their median values. As a result, probabilistic results for this case, presented in the top panel of Figure 5.1-18, show a considerable reduction in overall spread of total dose as compared to the reference case (Figure 4.1-5). Regression analysis results for this simulation, in terms of uncertainty importance factor as a function of time, are presented in the bottom panel of Figure 5.1-18. As expected, natural system parameters such as infiltration scenario and SZ groundwater flux emerge as key drivers of risk, with secondary contributions from a few of the waste package related parameters which were treated probabilistically. Note the consistency in importance ranking from this simulation and the classification and regression tree analysis discussed in the previous paragraph.

5.1.3 Igneous Scenario, Total Dose

Section 4.2 presents a discussion of the range of dose that the average member of the critical group is likely to receive over 100,000 years. As in the case of the nominal scenario, Figure 4.2-1 shows a broad range in projected dose rates at any given time. Unlike the nominal scenario, almost all of the realizations produced dose in excess of 10^{-5} mrem/yr. Regression analysis was carried out at time slices of 1,000, 10,000, and 100,000 years. Stepwise rank regression models between total dose and a set of statistically significant input variables were built for each case, and the most important variables were identified based on the value of uncertainty importance factor.

Figure 5.1-19 shows a bar chart of these results generated for the 1,000 -year data, indicating that the annual frequency of igneous intrusion is the most important variable, followed by wind speed. Similar importance analyses results for 10,000 years are shown in Figure 5.1-20, where the annual frequency of igneous intrusion is again the most important variable, followed by secondary contributions from time of igneous intrusion and wind speed. At 50,000 years, the dominant variable in terms of importance is the annual frequency of igneous intrusions, with SZ groundwater flux and infiltration scenario providing only marginal explanatory power for the overall spread in total dose (Figure 5.1-21).

An application of classification tree analysis to the same data sets to determine which variables control extreme outcomes for the igneous scenario is discussed next. As with the nominal scenario analysis, the dose values at any given time slice were first categorized as "high" if the values were in the top 10 percentile, or "low" if the values were in the bottom 10 percentile. The classification tree analysis algorithm then determined which variables were most capable of explaining the separation of these realizations into the appropriate categories.

Figure 5.1-22 (top) shows a decision tree summarizing the classification tree analysis in terms of the two most important variables for the 1,000-year data. In this case, much of the separation into high and low dose values can be explained on the basis of a single variable, the annual frequency of igneous intrusion. The second most important variable, wind speed, provides only marginal additional explanatory power. Figure 5.1-22 (bottom) is a partition plot of the same data, showing where the high and low dose producing clusters actually occur in the bivariate parameter space of the two most important variables.

It should be pointed out that although the igneous scenario calculations were carried out using 5,000 realizations (see section 4.2), the uncertainty importance analyses presented here are restricted to a random subset of 1,000 samples drawn from this population of 5,000 samples because of computational limitations in the regression analysis software. However, an analysis of the rank correlation coefficient between total dose and key uncertain inputs indicated excellent agreement between values derived from 1,000 and 5,000 sample data sets. Such agreement, both in terms of absolute magnitude and relative ordering of the input-output rank correlation coefficients, provide confidence in the regression analysis results based on the subset of 1,000 sample values.

5.1.4 Significance of Uncertainty Importance Analysis Results

To recapitulate, the objective of uncertainty importance analysis is to identify which variables affect the overall spread (variance) in total dose, as well as to identify which variables affect extreme outcomes in probabilistic model results. It is important to note that uncertainty importance, as defined in this section, is a function of: (a) sensitivity of the output to the input variable of interest, and (b) uncertainty of that input variable. In general, variables with high importance ranking satisfy both of these criteria. Conversely, variables which do not show up as important per these metrics are either restricted to a small range in the probabilistic analysis, and/or are variables to which the model outcome does not have a high sensitivity. Also, variables fixed at bounding/conservative values will not be identified as important because of the same reasons.

It should be stressed that uncertainty importance analysis results (as presented here) are conditional to the current TSPA-SR model and all of its underlying assumptions. Therefore, variables identified as important in previous TSPAs may not recur as key uncertainties in the present study, because:

- The conceptual model of a process has changed, and the overall outcome is not sensitive to underlying parameters in the model.
- The data base for a given parameter has been updated with a reduced uncertainty range.
- Variables treated as stochastic in previous TSPA iterations are fixed at conservative/bounding values, thus excluding them from the regression-based uncertainty importance analysis.

As such, uncertainty importance analysis results have to be considered as only part of the answer with respect to the significance of component models and uncertain parameters. A complete evaluation of such issues can be accomplished by combining uncertainty importance analysis with one-off sensitivity analysis, robustness analysis and barrier importance analysis—the results of which are presented in the following sections.

5.2 SENSITIVITY ANALYSES

The uncertainty importance analysis discussed in Section 5.1 shows that in the nominal scenario the most important part of the system is the waste package, whereas in the igneous disruption scenario the most important factor is the probability of having an igneous event. In this context, importance means having a significant impact on the uncertainty in the final calculated dose. Thus, waste package failure (represented by several parameters from the waste package degradation model) contributes strongly to the uncertainty in the nominal-scenario dose (Figure 5.1-4), and igneous-event probability is by far the biggest contributor to the uncertainty in the igneous-scenario dose (Figure 5.1-22). At later times, after most of the waste packages have failed, the natural system becomes more important in explaining the spread in the nominal-scenario dose results (Figure 5.1-11).

To further illustrate the effects of a number of the most important parameters on the TSPA results, analyses were performed in which parameters were fixed at particular values, or alternative assumptions were made. Such analyses demonstrate the effects of individual parameters or assumptions more explicitly than the uncertainty importance analysis can. These analyses are called "one-off" sensitivity analyses because changes are made to one parameter at a time (in some cases, changes are made to more than one parameter). Results of a number of them are presented in this section. In most cases, the sensitivity to individual parameters is examined by setting a parameter to its 5th and 95th percentile values. This choice keeps most of the range that is considered defensible. The 5th and 95th percentiles are used rather than the entire range (i.e., 0th and 100th percentiles) because in some cases there is a very long tail out to extremely unlikely parameter values. The 5th and 95th percentile values are at the level that they are unlikely, but not so unlikely as to be unreasonable. A few exceptions to the 5th and 95th "rule" are to be found in Section 5.2, specifically related to solubilities (Section 5.2.4) and UZ transport parameters (Section 5.2.6).

It should be emphasized here that the following uncertainty and sensitivity analyses are conditional on the current TSPA-SR models and assumptions. Further, some of the assumptions and models are conservative and therefore the uncertainty and sensitivity analyses have to be interpreted with care. That is, by restricting the range of the uncertain probability distributions, conservatism in model components can cause that particular model to show up as unimportant in either the regression analyses of Section 5.1 or in the sensitivity analyses of Sections 5.2 and 5.3. In particular, a specific model or parameter might have a great effect on performance if it were varied, but the assumed range of variation is narrow, so it shows up as unimportant in sensitivity and importance analyses. An example of a parameter with this effect is neptunium solubility (see Section 5.2.4.2). An example of a conceptual model that might have this effect is the dual-porosity UZ transport model, which may result in faster transport than a dual continuum model.

It is rather difficult *a priori* to quantify the degree of conservatism among the different component models. For coupled models, conservatism in one model can mask the importance of another model. Also, until dose consequences are estimated (assuming they represent the most appropriate metric), it is not possible to quantify the relative conservatism in one model or parameter versus another. For example, uncertain K_ds in the SZ are conservatively underestimated and therefore they do not show up as being important. Also, uncertainty analyses based on dose rate as the metric necessarily deal only with those radionuclides that pass through the potential repository system. Those that are retained, for example the majority of the uranium, cannot influence these types of analyses. Thus, a case can be made that the relatively immobile waste form itself (comprised mostly of uranium) is the most important part of the system, rather than the waste package. Also, as the models, assumptions, and uncertainties are refined, other parts of the system, such as the SZ, may become relatively more important. Seepage is another example of where the performance (and/or conservatism) of one submodel masks variations or sensitivity of the system to performance of another submodel. Specifically, because diffusive transport from the waste form is quite high in the TSPA-SR models and because the vast majority of packages are assumed to never encounter dripping conditions, variations in the seepage model, which only affects advective radionuclide transport, will not appear as very important to the potential repository performance. Appendix F gives a list of other conservatisms in the various models.

The sensitivity analyses in this section were performed using probabilistic TSPA simulations with 100 realizations. One hundred realizations were used rather than the 300 used for the analysis of the base case in Sections 4.1, 4.2, and 5.1 because 100 realizations are sufficient to see the relative effects when comparison is made to the 100-realization base-case simulation. (See Section 4.1.4 for a comparison of the 100-realization and 300-realization base cases.)

5.2.1 Unsaturated Zone Flow

5.2.1.1 Sensitivity to Infiltration

Section 5.1 indicates the most important UZ parameter to be the amount of infiltration at the surface, which also affects the amount of seepage that enters emplacement drifts, the thermal hydrologic environment in the drifts, and the radionuclide transport time through the UZ. Infiltration starts to have a significant influence on the nominal-scenario results at about

100,000 years (see Figure 5.1-3) and is very important to the nominal-scenario dose uncertainty at times after 100,000 years (see Figure 5.1-11).

The uncertainty in infiltration results from both the uncertainty in climate (precipitation and temperature) and the uncertainty in infiltration processes at the surface of the mountain. The amount of infiltration uncertainty included in the TSPA model is shown in Table 3.2-2, which gives the repository-averaged infiltration for the three infiltration cases (low, medium, and high) and the three climates (present-day, monsoon, and glacial-transition). The TSPA results for the nominal scenario do not depend on the present-day and monsoon climates because they occur only within the first 2,000 years, and all of the waste packages last at least 10,000 years in the nominal TSPA model. Thus, it is the uncertainty in the glacial-transition infiltration that is important to the nominal-scenario results. Note from Table 3.2-2 that the glacial-transition infiltration is more uncertain on the low side than on the high side. The low infiltration is almost a factor of 10 lower than the medium infiltration, while the high infiltration is only about a factor of 2 higher than the medium infiltration.

Two one-off analyses were performed in which the infiltration was fixed at its low and high values, rather than being sampled from a distribution. The results of these analyses are shown in Figure 5.2-1, which shows the mean dose curve from the nominal-scenario base case, along with the mean dose curves from the two sensitivity cases (infiltration fixed at low and infiltration fixed at high). As expected from the infiltration values in Table 3.2-2, there is much greater effect from the low case than from the high case. The "high" curve in Figure 5.2-1 is quite close to the "base case" curve, but the "low" curve is significantly lower. The larger effect of the "low" case results both from the greater difference in infiltration and because it has a lower probability weighting (see Table 3.2-2), so the low case is sampled less often than the others in the nominal model.

5.2.1.2 Sensitivity to Seepage Flow-Focusing Factor

The seepage abstraction model includes a parameter called the flow-focusing factor that represents the potential for channeling of flow on intermediate scales (see Section 3.2.4). It is of interest to examine the impact that this parameter has on the TSPA results, to determine whether it has enough impact on the results to warrant additional study of flow focusing above drifts. As shown in Table 3.2-3, the flow-focusing factor is represented in the TSPA model as a log-uniform probability distribution, with different limits on the distribution depending on the infiltration case (lower infiltration is associated with higher values of the focusing factor).

To examine the effects of this factor, analyses were performed with the flow-focusing factor fixed at its 5th and 95th percentile values, rather than being sampled from the distributions. It is not clear *a priori* what results to expect from these analyses. As noted in Section 3.2.4.3, flow focusing generally increases the seep flow rate for the locations that have seepage, but at the same time it decreases the fraction of locations that have seepage, with the total amount of seepage higher than it would have been without focusing. So, the net effect on calculated doses depends on whether the seepage fraction (fraction of waste packages with seepage) or the seep flow rate has more impact on radionuclide releases.

The results of the analyses are shown in Figure 5.2-2a, which shows the mean dose curve from the base case along with the mean dose curves from the two sensitivity cases (flow-focusing factor fixed at 5th percentile and 95th percentile values). The figure shows that the flow-focusing factor has no significant effect on calculated doses for about the first 40,000 years. The lack of effect at earlier times occurs because very little water is able to enter waste packages until about 40,000 years (see Figure 4.1-14). After 40,000 years, the higher flow-focusing factor results in higher doses than the lower flow-focusing factor, indicating that the total quantity of water is more important to dose than the number of waste packages affected. The differences between the two sensitivity cases and the base case is not very large, though, indicating that focusing of flow above the emplacement drifts is not particularly important to the TSPA results. The mean dose for the 95th-percentile case becomes approximately equal to the mean base-case dose after about 80,000 years.

Another sensitivity analysis to unravel the effects of the flow focussing factor and the effects of seepage spatial variability is shown in Figure 5.2-2b. In this case the flow focussing factor was set equal to 1.0, implying no flow focussing. Plus, the "local" seepage fraction is set to 1.0, which means that all of the 635 locations in the T-H model have the possibility of seeps if their percolation flux is high enough (greater than about 3.5 mm/yr). This effectively eliminates the seepage fraction variability within each of the 5 seepage environments. Specifically, for the glacial transition climate the result is that seepage environments 10-20 mm/yr, 20-60 mm/yr, and >60 mm/yr all have a "global" seepage fraction of nearly 100 percent (i.e., all packages are in the always drip environment), while seepage environment 3-10 mm/yr has about 70 percent of its packages in the always drip environment, and the 0-3 mm/yr seepage environment (which corresponds only to the low infiltration scenario) has 100 percent of its packages in the no-drip environment. This particular sensitivity case is similar to the base-case seepage model in the TSPA-VA.

As a further test of the effect of seepage, the seepage uncertainty factor was set equal 0.95 and combined with the sensitivity case described in the preceding paragraph. With the local seepage fraction set equal to 1.0, the effect of this is to set the mean "local" seepage flux to its 95th percentile value and the corresponding standard deviation to the 95th percentile value. These two values are used in a beta distribution that is sampled randomly over the 635 T-H locations, so there is still some spatial variability related to flux. This result is shown as the red curve in Figure 5.2-2b.

Because of the high diffusive releases from the waste packages, neither of the above two cases has a very large effect, particularly at early times, prior to 40,000 years, when the drip shield is mostly intact and ^{99}Tc dominates the dose rate. Later, as the drip shields fail and more and more patch openings occur in the waste packages, advective transport from the packages takes on a greater role in combination with the release of solubility-limited ^{237}Np. This is apparent in both sensitivities in Figure 5.2-2b, with the case having the 95th percentile local seepage flux showing a greater effect, as expected. An even larger effect might be observed if the seepage into the packages were not reduced by the ratio of patch opening area to total surface area of the waste package.

5.2.2 Engineered Barrier System Environments

Because most parameters related to the EBS environment, such as pH, RH, and temperature, are modeled deterministically rather than stochastically, no individual parameters relating to environments have been identified as very enlightening for one-off sensitivity analyses comparable to the one-off analyses in other subsections of Section 5.2, i.e., setting individual stochastic parameters to their 5th and 95th percentiles within the various submodels. However, the influence of various combinations of EBS environmental parameters is examined in Sections 5.3.4.2 and 5.3.5. Furthermore, the effect of the alternative repository design with backfill, discussed in Section 4.6, is mainly a function of the EBS environment, in particular, the temperature and RH on the waste package surface, at the cladding surface, and in the invert. Section 4.6 also contains an analysis of an alternate low thermal design whose main impact is to change the temperature and RH of the EBS environment. Little effect was found, partly because the waste package degradation model itself is insensitive to environmental parameters, such as pH and temperature, as described in the next section.

5.2.3 Waste Package and Drip Shield Degradation

This section reports sensitivity analyses of the nominal repository performance to a number of major parameters of waste package degradation processes. The sensitivity analysis results are analyzed in terms of the mean dose rate for the average member of the critical group (see Section 4.1 for details) and the mean waste package failure profile.

As for the base-case analysis (see Section 3.4), the WAPDEG model is used for the sensitivity analyses. The conceptual model and logic flow of the base case WAPDEG model (CRWMS M&O 1998 [145618]) are described in Section 3.4. The following simulation parameters used in the sensitivity analyses are the same as those for the base-case analysis (see Section 3.4).

- Temperature, RH, and contacting solution pH histories in the presence or absence of backfill
- 400 waste package and drip shield pairs
- 20 mm thick waste package outer barrier (Alloy-22)
- 15 mm thick drip shield (titanium)
- 1000 patches per waste package
- 500 patches per drip shield.

The sensitivity analysis cases analyzed in this section are:

- Sensitivity to the residual stress state uncertainty at closure-lid welds
- Sensitivity to the SCC model parameters for closure-lid welds

- Sensitivity to the Alloy-22 mean general corrosion rate
- Sensitivity to the uncertainty and variability partitioning ratio for Alloy-22 general corrosion rate.

Because temperature and RH do not significantly affect waste package and drip shield degradation, except for the RH threshold for corrosion initiation (see Section 3.4), a representative set of temperature and RH histories were used in these sensitivity analyses. Different waste types (i.e., CSNF waste packages, HLW, etc.) could give rise to differing thermal hydrologic conditions on the drip shields and waste packages. However, as stated above, drip shield and waste package degradation are not sensitive to RH and temperature conditions; therefore, no sensitivity analysis was conducted for different waste-type waste packages. In addition, the presence of drips is required for localized corrosion of the drip shield and the waste package outer barrier. However, the initiation threshold of the materials is much higher than the conditions expected in the potential repository (i.e., the corrosion potentials of the materials in the expected repository conditions are less than the critical corrosion potential of the materials). Hence, no localized corrosion is initiated in the WAPDEG analyses (see Section 3.4). Other corrosion models (general corrosion and SCC) are not dependent on dripping conditions (i.e., drip vs. no-drip); therefore, no sensitivity analysis was conducted for the effect of differing dripping conditions on waste package degradation.[1] As in the base-case analysis, potential performance credit for the stainless steel inner layer of the waste package was not considered in the sensitivity analyses. Details of the approaches and assumptions associated with the WAPDEG analyses are described in the supporting report, WAPDEG Analysis of Waste Package and Drip Shield Degradation (CRWMS M&O 2000 [146427]).

5.2.3.1 Sensitivity to Residual Stress State Uncertainty at Closure-Lid Welds

As discussed for the base-case analysis results, initial waste package failures are by SCC at the closure lid welds. Among the SCC model parameters considered in the analysis, the uncertainty ranges of the residual stress (hoop stress) and corresponding stress intensity factor are considered the most important. Sensitivity analyses were conducted to evaluate the effect of the residual hoop stress uncertainty (and corresponding stress intensity factor uncertainty) on the potential repository performance. The mean waste package failure profiles are also included in the analyses. For the continuity of the analyses of the results, the base case SCC model and the WAPDEG implementation are summarized in the following paragraph.

Because complete stress mitigation may not be possible for the closure-lid welds, the welds may be subject to SCC. Once a SCC crack initiates, it penetrates the closure-lid thickness in a very short time (see Section 3.4). Thus, stress mitigation in the closure-lid welds is a key design element to avoid premature failures of waste packages by SCC. In the slip dissolution model, the following two conditions should be met before initiating a SCC crack propagation in a

[1] This lack of dependence on dripping is probably a conservatism assumption in the case of SCC, since aggressive dripping water chemistry is unlikely to be present at all times on all packages because of two reasons: (1) the presence of the drip shield and (2) only a small fraction of the packages are in a drift seepage environment. On the other hand, dust is presumed to deposit ubiquitously on the package surfaces and this dust is assumed to be composed of NaNO3, which is the most aggressive chemical environment for inducing corrosion.

patch: (1) the stress intensity factor (K_I) should be positive, and (2) the stress state must be greater than or equal to the threshold stress. The presence of a compressive stress zone (or layer) in the outer surface delays the initiation of SCC. However, the compressive zone is slowly removed by general corrosion. The delay time depends on the compressive zone thickness and the general corrosion rate sampled for the patch. Details of the residual stress and stress intensity factor abstraction are discussed in the supporting report, *Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier and the Stainless Steel Structural Material* (CRWMS M&O 2000 [148375], Section 6.3).

In addition, all preexisting manufacturing defects in a patch, including embedded defects in the outer quarter of the thickness, are assumed to be surface breaking and oriented in the radial direction. This is a conservative assumption because many of the defects are likely to orient horizontally (i.e., in parallel to the weld line). Those cracks likely respond to the radial stress and, if SCC initiates, grow along the circumference of the closure-lid welds. The tip of all the manufacturing defects are assumed to advance at the general corrosion rate sampled for the patch. This is based on the modeling assumption that the same exposure condition that a patch experiences during a given time step is also applicable to the interior of defects in the patch. Growth of the preexisting defects at the general corrosion rate of the patch is a conservative assumption. Therefore, patches with preexisting defects would be subject to SCC earlier than other patches without defects. Details of the model implementation and assumptions are discussed in the supporting report, *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (CRWMS M&O 2000 [146427], Sections 5, 6).

Figure 5.2-3 illustrates the mean predicted dose rate when the residual hoop stress state at the closure-lid welds is fixed at the 95th and 5th percentile values of the uncertainty distribution described in Section 3.4. The results are compared with the base-case results. As expected, the mean dose is significantly affected when the residual stress-state changes, which affects the SCC failure of waste packages. The effect on waste package failure is shown in Figure 5.2-4. With the hoop residual stress fixed at the 5th percentile value, there is no SCC failure of waste package, (i.e., all the waste packages fail by general corrosion). This is consistent with the base-case results that SCC is the dominant waste package degradation process and an important process for the potential repository performance. In addition, the variance in the dose decreases significantly when this parameter is fixed at a discrete value.

As discussed in Section 3.4, the estimated long lifetime of the waste packages in the base case analysis is attributed mostly to the following two factors: (1) the stress mitigation to substantial depths in the dual closure-lid welds, which delays the onset of SCC crack propagation until the compressive zone layer is corroded; and (2) the very low general-corrosion rate applied to the closure-lid welds to corrode the compressive stress zones, which renders a long delay time before initiating SCC crack propagation. Substantial uncertainties are associated with the current SCC analyses, especially the stress mitigation on the closure-lid welds. The current assumption for the radial orientation of all the manufacturing defects in the closure-lid welds is conservative, because most embedded defects are likely to be oriented such a way that would not lead to radial cracks.

5.2.3.2 Sensitivity to Alternative Uncertainty Ranges of Major SCC Model Parameters for Closure-Lid Welds

As discussed in the previous section (Section 5.2.3.1), the major parameters in the SCC analysis of the waste package closure-lid welds are: (1) stress state and stress intensity factor, (2) threshold stress for SCC crack propagation, and (3) orientation and size of manufacturing defects. All three parameters are uncertain. Sensitivity analyses were conducted to evaluate the effects of those uncertain SCC model parameters on the potential repository performance. Four cases were evaluated in the current sensitivity analyses by changing one or more of the three parameters in each case. The four cases, along with the base case, are summarized in Table 5.2-1. For each case the parameter (or parameters) changed is indicated in bold.

Table 5.2-1 Summary of the Four Cases Evaluated in the Sensitivity Analyses for Alternative Uncertainty Ranges of the SCC Model Parameters for the Waste Package Closure-Lid Welds

Case	Residual Hoop Stress State Uncertainty	Threshold Stress Uncertainty	Manufacturing Defect Orientation
Base Case	- ± 30 percent of yield strength. - Symmetrical triangular distribution with the mode equal to 0	- 20 to 30 percent of yield strength. - Uniform distribution between 20 and 30 percent	- 100 percent defects with radial orientation (radial SCC crack propagation)
Case 1	- ± 30 percent of yield strength. - Symmetrical triangular distribution with the mode equal to 0	- 10 to 40 percent of yield strength. - Uniform distribution between 10 and 40 percent	- 100 percent defects with radial orientation (radial SCC crack propagation)
Case 2	- ± 10 percent of yield strength. - Symmetrical triangular distribution with the mode equal to 0	- 20 to 30 percent of yield strength. - Uniform distribution between 20 and 30 percent	- 100 percent defects with radial orientation (radial SCC crack propagation)
Case 3	- ± 10 percent of yield strength. - Symmetrical triangular distribution with the mode equal to 0	- 10 to 40 percent of yield strength. - Uniform distribution between 10 and 40 percent	- 100 percent defects with radial orientation (radial SCC crack propagation)
Case 4	- ± 10 percent of yield strength. - Symmetrical triangular distribution with the mode equal to 0	- 10 to 40 percent of yield strength. - Uniform distribution between 10 and 40 percent	- 1 percent defects with radial orientation (radial SCC crack propagation) - 99 percent defects with horizontal orientation (no radial SCC crack propagation)

The sensitivity analysis results for the mean dose and mean waste package failure profile are shown in Figures 5.2-5 and 5.2-6, respectively. Comparison of the base case with Case 1 and Case 2 with Case 3 shows that the threshold stress has insignificant effects on the mean dose rate and mean waste package failure. Comparison of the base case with Case 2 demonstrates that the uncertainty range of the residual hoop stress state (and corresponding stress intensity factor) has significant effects on the mean dose and mean waste package failure profile. Reduction in the residual hoop stress uncertainty range from ±30 percent to ±10 percent of the yield strength delays the first failure time of the mean waste package failure profile from about 12,000 years to about 20,000 years and shifts the failure profile curve to a substantially later time period. The mean dose rate curves are shifted accordingly (Figure 5.2-5).

Reduction in the number of manufacturing defects having a radial orientation by a factor of 100 (Case 4) significantly decreases the number of waste packages that fail by SCC (Figure 5.2-6) and delays the mean dose rate substantially (Figure 5.2-5). The initial failure time of the mean waste-package failure profile is delayed to about 32,000 years, and the mean dose rate is close to zero until about 40,000 years. For this case, most waste packages, except those that fail initially, fail by general corrosion.

These sensitivity analyses demonstrate that the uncertainty range of the residual hoop stress (and corresponding stress intensity factor) and the number (and size) of manufacturing defects having radial orientation are the two most important parameters that affect the SCC failure of waste packages and thus the potential repository performance.

5.2.3.3 Sensitivity to Uncertainty and Variability Partitioning Ratio for Alloy-22 General Corrosion Rate

This section and following section (Section 5.2.3.4) analyze the sensitivity of the potential repository (and waste package) performance to the parameters that are relevant to representing the uncertainty and variability of Alloy-22 general corrosion rate. As discussed in Section 3.4, the WAPDEG analysis yields an explicit representation of the uncertainty and variability in waste package (and drip shield) degradation. For the corrosion models and parameters for which data and analyses are available to quantify their uncertainty and variability, they were represented explicitly in the WAPDEG analysis. For other corrosion models and parameters for which uncertainty and variability are not quantifiable, the GVP technique was used to separate the variances due to uncertainty and variability from the total variance (see Section 3.4). In this technique, uncertainty is defined as the uncertainty in the mean value, and variability as the variance about that mean. In the analysis the fraction of the total variance to separate the uncertainty and variability variances was treated as an uncertain parameter and sampled randomly for each realization. The separation results in two distributions, one for the uncertainty and the other for the variability. Then the median of the variability distribution is sampled from the uncertainty variance, and the variability distribution is reconstructed around the sampled median.

In the WAPDEG analysis, variability in the waste package corrosion processes is represented by sampling the values of the individual corrosion model parameters from their variability distributions, and by populating the sampled values over the waste packages (referred to as package-to-package variability) and, if considered, the patches in a single waste package (referred to as patch-to-patch variability). Detailed discussions of the uncertainty and variability representation in the WAPDEG analysis are given in the supporting report, *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (CRWMS M&O 2000 [146427], Section 6).

The general corrosion rate distribution for Alloy-22 (waste package outer barrier) that was developed from the measurement data from the Project's Long-Term Corrosion Testing Facility is considered a mix of uncertainty and variability of the general corrosion rate. However, quantification of uncertainty and variability in the corrosion rate measurements is limited because the corrosion rates are extremely low and considered to be within the measurement noise. Because of this, it is difficult to quantify what the fraction of the total variance in the parent distribution represents the uncertainty and what fraction represents the variability. As discussed above, in the WAPDEG analysis, the fraction for the separation of the uncertainty and variability from the parent distribution is treated as an uncertain parameter.

Sensitivity analyses were conducted to evaluate the effect of the uncertainty-variability partitioning ratio by fixing the ratio at the 95th and 5th percentile values (i.e., using a 95 percent to 5 percent and a 5 percent to 95 percent uncertainty-variability partition, respectively). The first case is for the case that 95 percent of the total variance in the Alloy-22 general corrosion rate is due to uncertainty and 5 percent due to variability, and the second case is for the case that 5 percent of the total variance is due to uncertainty and 95 percent due to variability. For those two cases the median general corrosion rate for the variability distribution is sampled as an uncertain parameter as described above.

The results are shown in Figure 5.2-7 for the predicted mean dose rate profile and in Figure 5.2-8 for the mean waste package failure profile. As shown in the figures, there is no significant effect on the mean dose rate.

5.2.3.4 Sensitivity to Alloy-22 Median General Corrosion Rate

As discussed in the previous section, after separation of the variances due to uncertainty and variability from the total variance, the median of the variability distribution is sampled from the uncertainty variance, and the variability distribution is reconstructed around the sampled median. Another set of sensitivity analyses were conducted to evaluate the effect of the (sampled) median general corrosion rate of Alloy-22 on the mean dose rate and mean waste package failure profiles. The analyses were conducted by fixing the median general corrosion rate at the 95th and 5th percentile values of the uncertainty variance. In the analyses, the partitioning ratio for the uncertainty and variability separation was treated as an uncertain parameter and sampled for each realization.

The analysis results for the predicted mean dose rate and mean waste package failure profiles are shown in Figures 5.2-9 and 5.2-10, respectively. As shown in the figures, this parameter significantly affects the rate of degradation of waste packages and variability in waste package failures, and thus the predicted mean dose rate profile. Comparison of the current analysis results with the results from the previous section (sensitivity to the uncertainty-variability partitioning ratio) demonstrates that the median general corrosion rate is a more significant parameter to the potential repository performance than the partitioning ratio.

5.2.4 Waste Form Degradation and Mobilization

5.2.4.1 Sensitivity to Waste Type

The degradation rates of the CSNF waste matrix and HLW glass differ by only a factor of 2 using the conditions calculated for inside the waste package (Figures 3.5-15 and 3.5-19). Although the degradation rate of the DSNF is constant and is much greater, the mass of waste is much less. Hence, the CSNF release rate and the combined release rate of HLW glass and

DSNF waste matrix are about the same in the first 100,000 years, such that there is no sensitivity to the type of waste form or waste package on a per package basis. However, the total inventories of CSNF and co-disposed HLW and DSNF differ greatly, leading to a contribution to dose that is substantially different (Figure 5.2-11).

Furthermore, after 100,000 years, greater differences in release rates from the waste packages emerge. The inventory in the co-disposal packages is depleted, such that its release rate decreases. Additional perforation of cladding from localized cladding corrosion makes more radionuclides available from CSNF, so its release rate does not decrease, and parameters related to cladding become important (see Section 5.3.4.1).

5.2.4.2 Sensitivity to Secondary Mineral Phases

In this section we show some sensitivities outside the range of the base-case parameter distributions—in part, because neptunium solubility in the base-case model had variability but no uncertainty. It was only a function of environmental parameters, such as pH, which showed a narrow range of spatial variability across the potential repository.

The solubility limits presented in Section 3.5.5 were developed with several conservative features: 1) pure mineral phases are selected to control solubility, while in reality a radionuclide could be controlled by solid solution or co-precipitation; 2) when there are several possible solubility controlling minerals, the most soluble one is chosen, unless experiments say otherwise; and 3) sorption of radionuclides is neglected. The solubility limit models with these conservative features could overestimate the dose rate. To assess how much the calculated performance might improve by utilizing more realistic dissolved concentration models, a sensitivity calculation on element solubility is shown in this section.

The sensitivity analysis is based on high drip rate tests at Argonne National Laboratory (CRWMS M&O 2000 [153105], which are a set of experiments simulating spent fuel dissolution under potential repository conditions. As discussed in Section 3.5.5, the solubility limits based on the Argonne experiments (as interpreted for this sensitivity study) are much lower than the base-case solubilities for two key dose contributors, Np and Th (CRWMS M&O 2000 [153105]. The Argonne-based solubility for Np is more than 3 orders of magnitude lower than the base-case solubility (in CSNF packages); the Argonne-based solubility for Th is more than 7 orders of magnitude lower; the Argonne-based solubility for Am is about 2 orders of magnitude lower (in CSNF packages); the Argonne-based solubility for U is about the same (in CSNF packages); and the Argonne-based solubility for Pu is about the same.

Figure 5.2-12 shows the results of using the secondary-phase-controlled solubility limits. The reduced solubilities have their greatest impact on late time doses when solubility-limited radionuclides, such as Np, begin to control the total dose rate (see Figure 4.1-5). Figure 5.2-12a indicates an approximately one order-of-magnitude reduction in the peak of the mean total dose rate history (at 100,000 years). Figure 5.2-12b indicates that most of the reduction in the 100,000-year total dose rate compared to the base case (see Figures 4.1-5, 4.1-6, and 4.1-7) is due the reduction in the ^{237}Np dose rate. For this secondary phase analysis, the major dose rate contributors in 100,000 years are ^{99}Tc, ^{129}I, ^{239}Pu, ^{227}Ac, and ^{231}Pa,and most of the contribution to the ^{239}Pu dose is from the irreversibly sorbed fraction of ^{239}Pu.

5.2.5 Engineered Barrier System Transport

5.2.5.1 Sensitivity to Invert Diffusion

For nominal performance, the dose rate at early times is sensitive to the form of the diffusion model (and therefore the diffusion rate) chosen for diffusion of radionuclides in the invert. Changes among diffusion models result in differences of several thousand years. At times greater than 30,000 years, the differences are small, and the peak dose rate at 100,000 years is relatively insensitive to the diffusion model (i.e., diffusion rate). Those conclusions are illustrated in Figure 5.2-13. The differences in the three cases shown in the figure are described in the following paragraphs.

For the base case, the mean value of the diffusion coefficient as a function of liquid saturation and porosity is given by:

$$D_{mean} = D_{fw} s^{1.849} \phi^{1.3} \qquad \text{(Eq. 5.2-1)}$$

where D is the diffusivity (cm^2/s), D_{fw} is the free-water diffusion coefficient (cm^2/s), s is the liquid saturation, and ϕ is the porosity. The value for D_{fw} ($= 2.299 \times 10^{-5}$ cm^2/s at 25°C) is a bounding value for all radionuclides of interest for the TSPA. The exponent on the saturation, s, is based on a statistical analysis of data from Conca and Wright (CRWMS M&O 2000 [150418]; CRWMS M&O 2000 [150792]). This value is slightly less than 2, which is the typical value for Archie's law in a fine sand. The exponent on the porosity is 1.3, the typical value for Archie's law. In fact, the statistical fit to moisture content justifies using an exponent of 1.849 for the porosity; however, it has conservatively been left at the typical value for Archie's law of 1.3.

For the high-diffusion case, the mean response of the diffusion coefficient is given by:

$$D = D_{fw} \phi^{1.0} s^{1.0} \qquad \text{(Eq. 5.2-2)}$$

For the low-diffusion case, the diffusion coefficient is constant at $D = 10^{-11}$ cm^2/s.

5.2.6 Unsaturated Zone Transport

5.2.6.1 Sensitivity to Matrix Diffusion

Figure 5.2-14 shows the mean dose rate from the base case compared to a case with no matrix diffusion in the UZ and also compared to a case where the UZ anion and cation matrix diffusion coefficients were set at 100 times the matrix diffusion coefficients in the base case. It should be noted that these parameter values are outside the range of base-case probability distributions, in contradistinction to most of the other analyses in Section 5.2.

The results show that UZ matrix diffusion has a moderate effect on the dose history, especially between 20,000 and 30,000 years. This "bump" in the dose curve for the no matrix diffusion case is caused by ^{243}Am, which has a half life of ~7,000 years. With matrix diffusion, the travel time of ^{243}Am through the UZ is long enough for ^{243}Am to decay, thereby attenuating the dose. Without matrix diffusion, the ^{243}Am can transport through the UZ without significant retardation or decay, causing the "bump" in the dose curve. The case with 100 times the base case matrix diffusion also implies that the base case probability distributions for matrix diffusion coefficients maximize the impact of matrix diffusion in the UZ, i.e., the best estimate ranges for matrix diffusion, based on available data, imply that matrix diffusion is an important phenomenon for transport in the UZ.

5.2.7 Saturated Zone Flow and Transport

The SZ flow and transport component of TSPA-SR is modeled independently of the TSPA calculations (Section 3.8). The modeling is incorporated in the TSPA calculations through a library of breakthrough curves. This structure would require a new library for every sensitivity case, and is therefore incompatible with one-off sensitivity analyses. Analyses examining a degraded SZ barrier, for which new breakthrough-curve libraries were created, are described in Section 5.3.7.

5.2.8 Biosphere

Two sensitivity analyses have been conducted to examine the range in nominal-scenario dose results due to the biosphere component of TSPA-SR. The sensitivity analyses address how the TSPA dose calculation is influenced by uncertainties in the BDCFs and uncertainties in the estimate of the water volume used by the proposed farming community. An analysis of uncertainties and sensitivities internal to the biosphere model is presented in Section 3.9.2.5.

5.2.8.1 Sensitivity to Biosphere Dose Conversion Factors

This sensitivity study addresses how the dose calculation is affected by the spread in the distributions used to define the BDCFs. BDCFs are the radionuclide-dependent factors used to convert radionuclide concentrations in groundwater into the annual dose incurred by a receptor within the critical group living 20 km from Yucca Mountain. Sensitivity to volcanic BDCFs are not considered here.

Figure 5.2-15 shows the mean value of the base-case nominal doses compared with curves labeled as 5th and 95th. These curves depict the mean-value results of the probabilistic TSPA base case (100 realizations) calculated with the values of the BDCFs held constant. In one case, the BDCFs for all the radionuclides are fixed at the 5th percentile value of their distributions and, for the other curve, at the 95th percentile value of their distributions. For any given radionuclide, the dose is proportional to the BDCF and for all the radionuclides the sum of the doses is proportional to a weighted sum of the BDCFs (weighted by radionuclide release rate). The curves have the same shape because the radionuclide release rates are the same in all three cases.

As shown in the figure, the BDCFs have little effect on the dose calculation. The 5th percentile curve reduces the mean dose by about half; the 95th percentile BDCFs increase the mean dose by approximately a quarter. As discussed in Section 3.9, most of the biosphere is prescribed by regulation. Using the average member of the critical group implies that the receptor has the average food consumption rates; using constant values for the consumption rates greatly reduces the spread in the BDCF distributions; little spread in the BDCF distributions implies little impact on the variance in the calculated doses. Because of the prescribed elements of the biosphere, the major sources of uncertainty in the BDCFs involve soil and dust pathways. The most important pathway in the nominal biosphere model is the drinking water pathway, and the drinking water rate is set to a constant.

5.2.8.2 Sensitivity to Water-Usage Volume

This sensitivity study addresses how the dose calculation is affected by the spread in the distributions used to define the water-usage volume. The water-usage volume is the estimated amount of water used in one year by a hypothetical farming community living 20 km from Yucca Mountain. The water-usage volume is the calculational basis for determining the radionuclide concentration in groundwater; i.e., it is the amount of water that contains all radionuclides released from a potential repository at Yucca Mountain and that cross the 20 km distance to the hypothetical farming community each year.

Figure 5.2-16 shows the mean value of the base-case doses compared with curves labeled as 5th and 95th. The 5th and 95th curves depict the mean-value results of the probabilistic TSPA base case (100 realizations) calculated with the values of the water-usage volume held constant; in one case, the water-usage volume is fixed at the 5th percentile value of the distribution and, for the other curve, at the 95th percentile value of the distribution. For any given radionuclide, the dose is inversely proportional to the water-usage volume; therefore, the curves have the same shape.

As shown in the figure, the water-usage volume has little effect on the dose calculation. The 5th percentile of the water-usage volume is 1.404×10^6 m^3/yr; the 95th percentile is 3.589×10^6 m^3/yr; the mean is 2.394×10^6 m^3/yr. The 5th percentile curve reduces the mean dose by about half; the 95th percentile BDCFs increase the mean dose by approximately half. The implication is that the dose calculation is relatively insensitive to the range of water-usage volume assumed in TSPA-SR.

5.2.9 Disruptive Events

As Section 3.10 describes, igneous disruption is the only disruptive scenario that has been identified as requiring explicit analysis in the TSPA. Section 4.2 describes the TSPA-SR results for the igneous disruption scenario, and Section 5.1.2 describes uncertainty importance analyses associated with the TSPA-SR results. This section presents the results of additional analyses that examine specific cases designed to test the robustness of the TSPA-SR results to alternative modeling assumptions. Results are presented in the context of one-off sensitivity analyses, in which selected parameter values are fixed and sampled values are used for all other parameters, or as comparisons of performance using alternative conceptual models. For analyses that compare results using fixed values for selected parameters, values are chosen representing the 5th and 95th percentiles of the distributions used in the TSPA-SR. Analyses of alternative assumptions about parameter values that only affect the eruptive release (e.g., changes in wind speed) have been done using only the eruptive pathway portion of the GoldSim model, to allow a clearer display of the changes in performance. All analyses in which the eruptive probability-weighted mean annual dose rate from the TSPA-SR are based on 100 realizations of 100,000 year performance using the same sampled values for all other parameters. All analyses that include calculation of dose rates from igneous intrusion groundwater release scenario are based on 1,000 realizations of 20,000-year performance, using the same set of sampled parameter values. As described in Section 4.2.3, the choice of 1000 realizations of 20,000-year performance provides adequate statistical coverage of during the period of greatest interest within reasonable computational constraints. The alternative modeling assumptions represented by these analyses are not considered to be realistic or appropriate for comparison to the proposed regulatory standards, and the probability-weighted mean 50,000-year dose rate described in Section 4.2.2 should be interpreted as the best estimate of future performance for the igneous disruption scenario class. Where shown, the 5th and 95th percentile dose rates represent extreme favorable and unfavorable deviations from expected performance that are equally likely (or unlikely) to occur.

5.2.9.1 Sensitivity to Alternative Models for the Probability of Igneous Activity

Figure 5.2-17 shows a comparison of the probability-weighted 20,000-year mean annual igneous intrusion dose rates, as described in Section 4.2.3, with the same dose rate calculated using a fixed annual probability of igneous intrusion equal to 10^{-7}, rather than a value sampled from a distribution with a mean of 1.6×10^{-8}. The conditional probability that an eruptive conduit intersects waste if an intrusion occurs within the repository footprint is set to 1, yielding an annual probability of 10^{-7} for both intrusive and eruptive events. This higher probability falls within the range of values sampled in the TSPA-SR (near the upper limit, at approximately the 99.5 percentile of the distribution sampled for igneous intrusion probability), and is the value used by the U.S. Nuclear Regulatory Commission (NRC) in analyses reported in the "Issue Resolution Status Report (Key Technical Issue: Igneous Activity, Revision 2)" (Reamer 1999 [119693], p. 10). The DOE recognizes that the value of 10^{-7} provides a useful upper bound to values that should be reasonably considered in the site recommendation. Because the event probability is used directly in the weighting of probabilistic doses, changes in event probability should result in a linear scaling of the mean annual dose. Figure 5.2-17 confirms this observation. The mean dose calculated using the fixed higher probability is about 17 times higher during the first 2,000 years than the mean dose calculated using the full distribution of probabilities for igneous intrusion and a conditional eruption probability of 0.36, as described in Section 3.10.1. At later times, the scaling between the curves varies slightly with time, reflecting both the sampling of the time of intrusion and the influence of individual realizations with varying probabilities on the location of the mean at different times. The highest peak mean dose is increased by approximately a factor of 10, to about 0.9 mrem/yr, and occurs at the end of the simulation, 20,000 years after closure.

5.2.9.2 Sensitivity to Assumption that the Wind Direction is Fixed toward the Location of the Critical Group

As described in Section 4.2, the TSPA-SR analysis of eruptive releases from igneous disruption includes an assumption that the wind direction is fixed in all realizations toward the critical group. This assumption is unrealistic given the wind data for the area, but it provides a conservative bound to uncertainty regarding wind direction. As discussed in Section 3.10.4, this assumption also compensates for uncertainty regarding the possibility that contaminated ash deposited by winds blowing in directions other than south might later be redistributed by wind or water to the location of the reasonably maximally exposed individual and could therefore contribute to the total probability-weighted dose from volcanic eruption.

Figure 5.2-18 shows a comparison of the probability-weighted mean annual igneous dose rate calculated using a fixed wind direction toward the critical group with the same dose rate calculated with the wind direction sampled from a wind rose based on available information, as described in Section 3.10. When the wind direction is sampled, dose is reduced by a factor of about 5 during the first approximately 2,000 years of the simulations, during the period when the annual dose rate is dominated by the eruptive release. This reduced dose rate is not presented here as a preferred alternative to the dose rate calculated assuming a fixed wind direction. The fixed-wind-direction analysis remains the preferred performance measure for the TSPA-SR because it conservatively bounds uncertainty related to future wind direction and compensates for uncertainty regarding surficial redistribution of contaminated ash and soil. Results of the comparison provide insight into the limits to potential reductions in the calculated dose rate that might be achieved by the development of realistic models for soil and ash redistribution. The extent to which such models might result in reduced annual dose rates during the first several thousand years after closure is unknown, but it will be no greater than the factor of approximately 5 shown here.

5.2.9.3 Sensitivity to Wind Speed

Figure 5.2-19 shows a comparison of the probability-weighted mean annual eruptive dose rate, as described in Section 4.2, with the same dose rate calculated using the 5th and 95th percentile value for the wind speed. The annual eruptive dose rate increases by a factor of 2 for the 95th percentile wind speed. The 95th percentile wind speed corresponds to a wind speed of 13.9 meters/second, and exceeds the value of 12 meters/second suggested by the NRC in the "Igneous Activity Issue Resolution Status Report" (Reamer 1999 [119693], p. 88) as a "reasonably-conservative basis to model aerial tephra dispersal." Dose rates increase by roughly a factor of 2 when wind speed is increased to the 95th percentile, and decrease by more than an order of magnitude at the 5th percentile.

5.2.9.4 Sensitivity to the Removal of Contaminated Soil by Erosion

As described in Section 3.10.3.2, the TSPA-SR model for the behavior of volcanic ash layers in the biosphere includes a characterization of soil removal processes. Soil is assumed to be eroded at a rate of 0.06 to 0.08 cm/yr, which corresponds to a mean erosion time for a 15 cm soil layer of 250 to 188 years. As discussed in Section 3.10, these values are appropriate for agricultural land that is consistent with the high air mass loading values assumed in the construction of the volcanic biosphere dose conversion factors. The analysis reported here uses an erosion rate that is set to 0.015 cm/yr, which corresponds to a mean erosion time for a 15 cm soil layer of 1,000 years. This assumption is unrealistic, but provides a conservative upper bound to doses that would result from alternative assumptions about soil removal rates. Note, for example, that the soil removal rates used by the NRC in analyses reported in their "Igneous Activity Issue Resolution Status Report" correspond to mean ash-layer erosion times ranging from 100 to 1,000 years (Reamer 1999 [119693], p. 14), with a preferred value of 1,000 years (Reamer 1999 [119693], p. 11).

Figure 5.2-20 shows that a 1,000-year mean soil erosion time results in probability-weighted annual eruptive doses that are a factor of 5 higher than the base case at 10,000 years. The probability-weighted annual eruptive doses for both cases reach a maximum level in the first

1,000 years and then decrease steadily. Eruptive doses for this assumption remain well below levels of concern, reaching a peak of about 0.01 mrem/yr.

5.2.9.5 Sensitivity to the Volume of Material Erupted

As described in Section 3.10, the ASHPLUME version 1.4LV code (CRWMS M&O 1999 [150744]) used in the TSPA-SR to simulate effects of volcanic eruptions uses the total volume of erupted material as the independent variable that defines the energy of the eruption. For example, the duration of the eruption and the height of the erupted column are derived parameters that are calculated within each realization based on the sampled value for erupted volume.

Figure 5.2-21 shows a comparison of the probability-weighted mean annual eruptive dose rate, as described in Section 4.2, with the same dose rate calculated using the 5th and 95th percentile value for the volume of material erupted. These percentiles correspond to erupted volumes of 0.0026 km^3 and 0.336 km^3, respectively, and correspond approximately to calculated column heights of 2 and 5 km above the ground surface. The total annual igneous dose rate is shown to be insensitive to range of values selected for erupted volume in this analysis, and is therefore insensitive to uncertainty regarding the energy of the eruptive event and the height of the eruptive column, both of which are derived from volume in ASHPLUME version 1.4LV (CRWMS M&O 1999 [150744]).

5.2.9.6 Sensitivity to the Number of Waste Packages Damaged by Volcanic Eruption

Figure 5.2-22 shows a comparison of the probability-weighted mean annual eruptive dose rate, as described in Section 4.2, with the same dose rate calculated using the 5th and 95th percentile values from the TSPA-SR distributions for the number of waste packages damaged during a volcanic eruption. These percentiles correspond to 6 and 16 waste packages, respectively. This comparison provides insight into the sensitivity of overall performance to uncertainty about diameter of eruptive conduits, their location, and the number of eruptive conduits within the potential repository associated with each igneous event. Results of this comparison show that performance is moderately sensitive to the total number of packages that are damaged during the eruptive event, and that peak dose may be increased by a factor of 1.5. However, overall peak mean dose from the eruptive event remains below 0.01 mrem/yr.

5.2.9.7 Sensitivity to the Number of Waste Packages Damaged by Intrusion

Figure 5.2-23 shows a comparison of the probability-weighted 20,000-year mean annual total igneous dose rate, as described in Section 4.2.3, with the same dose rate calculated using the 5th and 95th percentile values from the TSPA-SR distributions for the number of waste packages damaged by igneous intrusion. As described in Section 3.10.2.3.2, damage to three packages on either side of the intrusive dike and one package intersected by the dike is assumed to be sufficient that the packages provide no further protection to the waste package. This region is referred to as "Zone 1." In the remaining portion of all drifts intersected by a dike, damage is limited to failure of lid welds due to high temperature and pressure. This region is referred to as "Zone 2." As described in the following section, damage in Zone 2 is limited to the formation of an aperture of uncertain (and sampled) cross-sectional area in each damaged package. The 5th and 95th percentile values for the number of packages damaged in Zone 1 are 108 and 219, respectively. The same numbers for Zones 1 and 2 combined are 132 and 6,516, respectively. For the purposes of this comparison, sampled values from the base case are used for the cross-sectional area of the aperture in Zone 2 packages.

This comparison provides insight into the sensitivity of overall performance to uncertainty about the potential repository's response to igneous intrusion. Results of this comparison show that performance is only moderately sensitive to the total number of packages that are damaged by intrusion, with peak dose increasing by less than a factor of 2. Overall peak mean dose from igneous activity rises only slightly above 0.1 mrem/yr for the 95th percentile case.

5.2.9.8 Sensitivity to the Magnitude of Damage to Packages in Zone 2

Figure 5.2-24 shows the probability-weighted 20,000-year mean annual total igneous dose rate, as described in Section 4.2.3, using the 5th and 95th percentile values from the TSPA-SR distributions for the diameter of apertures in waste package lids damaged by intrusion and the number of waste packages damaged by igneous intrusion. Values for the 5th and 95th percentile apertures are 3.5 cm^2 and 30 cm^2, respectively. This result compared with the mean curve from Figure 5.2-23 shows that performance is insensitive to the range of diameters considered in the analysis for the apertures in Zone 2 waste packages damaged by intrusion.

5.2.9.9 Sensitivity to Potential Doses during the Eruptive Event

This section examines the potential radiation dose received by an average member of the critical group who does not leave the region during the eruptive event. Doses are calculated using the eruptive-phase BDCFs described in Section 3.10.3.1 (DTN: MO0006SPAPVE03.001 [151768]) and the radionuclide inventory at year one. The analysis assumes PM_{10} (particles less than 10 microns) air mass loading of 1000 $\mu g/m^3$, as described in Section 3.10.1, and a uses a median eruptive duration of 8.2 days (CRWMS M&O 2000 [142657], Table 5). As described below, results indicate that the probability-weighted annual dose from this pathway will be on the order of 10^{-3} to 10^{-4} mrem/yr, significantly below the probability-weighted eruptive dose calculated for long-term exposure to the contaminated ash layer.

The dose for each radionuclide is calculated from the following equation (DTN: MO0006SPAPVE03.001 [151768]):

$$\text{Dose} = PM_{10} \times C_{ash} \times \text{BDCF} \qquad \text{(Eq. 5.2-3)}$$

where

- Dose = the dose from each individual radionuclide
- PM_{10} = the mass concentration of PM_{10} fraction of suspended particulates (g/m^3)
- C_{ash} = the activity concentration of radionuclide in ash (pCi/g ash)
- BDCF = the biosphere dose conversion factor for each individual radionuclide ($rem/day/pCi/m^3$).

This equation requires calculating the dose from each radionuclide separately and then summing the resulting individual doses to obtain the total dose to the average member of the critical group during the eruptive phase of the volcanic event. This summation of total dose is:

$$\text{Total Dose} = \text{summation (dose from each radionuclide)} \qquad \text{(Eq. 5.2-4)}$$

The 17 radionuclides for which eruptive phase BDCFs were provided are shown in Table 3.10-6, and are repeated here in Table 5.2-2 for convenience.

Table 5.2-2. BDCFs for the 17 Radionuclides Relevant to the Volcanic Event

Radionuclide	BDCF (rem/day/pCi/m^3)
^{227}Ac	6.92e-02
^{241}Am	4.64e-03
^{243}Am	4.60e-03
^{137}Cs	1.36e-06
^{231}Pa	1.34e-02
^{210}Pb	2.07e-04
^{238}Pu	4.10e-03
^{239}Pu	4.49e-03
^{240}Pu	4.49e-03
^{242}Pu	4.29e-03
^{226}Ra	1.16e-04
^{90}Sr	5.39e-06
^{229}Th	2.22e-02
^{230}Th	3.36e-03
^{232}U	6.80e-03
^{233}U	1.40e-03
^{234}U	1.37e-03

Source: DTN: MO0006SPAPVE03.001 [151768]

The activity concentration (C_{ash}) of radionuclides in ash is calculated separately for each radionuclide. This is done by first determining the mean value of the grams of fuel per gram of ash for all radionuclides combined together. This value was determined by extracting the average g/cm^2 of fuel and average g/cm^2 of ash that fell on the critical group 20 km downwind (assuming the wind always blew towards the critical group) from 100 simulations of the base case ASHPLUME version 1.4LV model (CRWMS M&O 1999 [150744]). The resulting means for fuel and ash concentrations were 2.77×10^{-6} g/ cm^2 and 2.30 g/cm^2, respectively. This results in a mean grams of fuel per gram of ash at the critical group of 1.20×10^{-6}. Put another way, each gram of ash that is deposited at the critical group contains 1.20×10^{-6} grams of the 17 radionuclides combined. These radionuclides are not present in equal quantities, and thus it is necessary to calculate the grams of each radionuclide per gram of ash and to then normalize these results so that the 17 radionuclides being tracked sum up to 1.20×10^{-6} grams for each gram of ash. This will ensure consistency with the ASHPLUME model results (CRWMS M&O 1999 [150744]).

The mass of each radionuclide in both CSNF and DSNF waste packages is listed in Table 5.2-3. The grouping of these into a hypothetical combined waste package is also shown assuming that 70 percent of the packages are CSNF and 30 percent are DSNF.

Table 5.2-3. Mass of Radionuclides in Waste Packages

Radionuclide	CSNF Mass	DSNF Mass	Combined Weighted Mass
^{227}Ac	3.09e-06	1.05e-04	3.37e-05
^{241}Am	8.76e+03	7.87e+01	6.16e+03
^{243}Am	1.29e+03	1.68e+00	9.04e+02
^{137}Cs	5.34e+03	5.52e+02	3.90e+03
^{231}Pa	9.87e-03	3.02e-01	9.75e-02
^{210}Pb	0	1.38e-08	4.14e-09
^{238}Pu	1.51e+03	8.79e+01	1.08e+03
^{239}Pu	4.38e+04	2.13e+03	3.13e+04
^{240}Pu	2.09e+04	4.55e+02	1.48e+04
^{242}Pu	5.41e+03	1.15e+01	3.79e+03
^{226}Ra	0	2.21e-06	6.63e-07
^{90}Sr	2.24e+03	3.01e+02	1.66e+03
229Th	0	2.46e-02	7.38e-03
230Th	1.84e-01	1.75e-02	1.34e-01
^{232}U	1.01e-02	1.37e-01	4.82e-02
^{233}U	7.00e-02	1.98e+02	5.94e+01
^{234}U	1.83e+03	2.77e+02	1.36e+03

Source: DTN: SN0003T0810599.010 [151021]

The total mass of these 17 radionuclides in this combined package is 6.50×10^4 g. This is well below the actual mass of a waste package because it does not include the mass of radionuclides (primarily ^{238}U) that are not significant contributors to dose.

The next step in the calculation is to take the mean grams of fuel per gram of ash for the combined radionuclides (1.20×10^{-6}) and multiply the percentage of the total package mass attributable to each radionuclide by this value. This results in the grams of each radionuclide per gram of ash and all 17 of these values sum to 1.20×10^{-6}. These results are shown in Table 5.2-4 below along with the specific activity for each radionuclide in Ci/g needed in the next step.

Table 5.2-4. Radionuclide Activities and Grams of Each Radionuclide per Gram of Ash

Radionuclide	Combined Weighted Mass	Percent of Total Mass	Grams Radionuclide per gram Ash	Specific Activity (Ci/gram-radionuclide)
^{227}Ac	3.37e-05	5.19e-08 %	6.22e-16	72.341
^{241}Am	6.16e+03	9.48 %	1.14e-07	3.4322
^{243}Am	9.04e+02	1.39 %	1.67e-08	0.19962
^{137}Cs	3.90e+03	6.00 %	7.21e-08	86.121

^{231}Pa	9.75e-02	1.50e-4 %	1.80e-12	0.047618
^{210}Pb	4.14e-09	6.37e-12 %	7.64e-20	75.326
^{238}Pu	1.08e+03	1.66 %	2.00e-08	17.12
^{239}Pu	3.13e+04	48.15 %	5.78e-07	0.062041
^{240}Pu	1.48e+04	22.77 %	2.73e-07	0.22787
^{242}Pu	3.79e+03	5.83 %	7.00e-07	0.003929
^{226}Ra	6.63e-07	1.02e-9 %	1.22e-17	0.98927
^{90}Sr	1.66e+03	2.55 %	3.06e-08	136.5
^{229}Th	7.38e-03	1.14e-5 %	1.36e-13	0.19761
^{230}Th	1.34e-01	2.06e-4 %	2.48e-12	0.020614
^{232}U	4.82e-02	7.42e-5 %	8.90e-13	22.365
^{233}U	5.94e+01	0.09 %	1.10e-09	0.010169
^{234}U	1.36e+03	2.09 %	2.52e-08	0.006236

The next step is to calculate the pCi of each radionuclide per gram of ash (C_{ash} in Eq. 5.2-3). This is done by multiplying the grams of radionuclides per gram of ash in the table above for each radionuclide separately by the activity for that radionuclide and then converting the result into pCi from Ci. The resulting values for C_{ash} are shown in Table 5.2-5 below along with the BDCFs for each radionuclide (Table 5.2-2).

Table 5.2-5. pCi of Each Radionuclide per Gram of Ash and BDCFs

Radionuclide	**pCi Radionuclide per gram Ash (C_{ash})**	**BDCF (rem/day/pCi/m^3)**
^{227}Ac	4.50e-02	6.92e-02
^{241}Am	3.90e+05	4.64e-03
^{243}Am	3.33e+03	4.60e-03
^{137}Cs	6.21e+06	1.36e-06
^{231}Pa	8.57e-02	1.34e-02
^{210}Pb	5.76e-06	2.07e-04
^{238}Pu	3.42e+05	4.10e-03
^{239}Pu	3.59e+04	4.49e-03
^{240}Pu	6.21e+04	4.49e-03
^{242}Pu	2.75e+02	4.29e-03
^{226}Ra	1.21e-05	1.16e-04

Table 5.2-5. pCi of Each Radionuclide per Gram of Ash and BDCFs (Continued)

Radionuclide	pCi Radionuclide per gram Ash (C_{ash})	BDCF (rem/day/pCi/m^3)
^{90}Sr	4.18e+06	5.39e-06
^{229}Th	2.69e-02	2.22e-02
^{230}Th	5.10e-02	3.36e-03
^{232}U	1.99e+01	6.80e-03
^{233}U	1.12e+01	1.40e-03
^{234}U	1.57e+02	1.37e-03

The final step is to calculate the dose for each radionuclide separately, using Eq. 5.2-3, and then sum the doses (Eq. 5.2-4) to get a total dose per day for the eruptive phase of the volcanic eruption. The doses are shown in the Table 5.2-6.

Table 5.2-6. Doses for Each Radionuclide During the Eruptive Phase of the Volcanic Event

Radionuclide	Dose (rem/day)
^{227}Ac	3.11e-06
^{241}Am	1.81
^{243}Am	0.015
^{137}Cs	0.008
^{231}Pa	1.15e-06
^{210}Pb	1.19e-12
^{238}Pu	1.40
^{239}Pu	0.161
^{240}Pu	0.279
^{242}Pu	0.001
^{226}Ra	1.40e-12
^{90}Sr	0.023
229Th	5.98e-07
230Th	1.71e-07
^{232}U	1.35e-04
^{233}U	1.56e-05
^{234}U	2.15e-04
Total Summed	**3.70**

The total dose to the average member of the critical group during the eruptive phase of a volcanic event that occurs in year one is 3.70 rem/day. Assuming this event lasts 8.2 days yields a total dose of 30.3 rem. The resulting probability weighted dose would be 4.8×10^{-4} mrem/yr, which is well below the peak mean annual dose for the base igneous case. Doses from eruptive events occurring at all times later than year one would be lower due to radioactive decay.

5.2.9.10 Sensitivity to Alternative Models for BDCFs

As discussed in Section 3.10.3.3, BDCFs have been developed (DTN: MO0006SPAPVE03.001 [151768]) for several different conditions that might exist following an eruptive event (Table 5.2-7).

Table 5.2-7. BDCFs for the Volcanic Igneous Event and the Effect on Dose (Normalized to the Results for Transition Phase [≤1cm] Average Member of the Critical Group BCDFs)

BDCF Description	Applicable Time Frame	Source Group Modeled	Relative Mean BDCF Effect on Dose
Transition Phase for Thin Ash Deposits (≤1cm)	For Several Years Immediately After Eruptive Phase Ends	Average Member of the Critical Group (AMCG)	100 percent (Set by Normalization)
Transition Phase for Thick Ash Deposits (15cm)	For Several Years Immediately After Eruptive Phase Ends	AMCG	63percent of Transition Phase (≤1cm) for AMCG
Steady State Phase for Thin Ash Deposits (≤1cm)	After Transition Phase Ends	AMCG	43percent of Transition Phase (≤1cm) for AMCG
Steady State Phase for Thick Ash Deposits (15cm)	After Transition Phase Ends	AMCG	60percent of Transition Phase (≤1cm) for AMCG
Transition Phase for Thin Ash Deposits (≤1cm)	For Several Years Immediately After Eruptive Phase Ends	RMEI	79percent of Transition Phase (≤1cm) for AMCG
Transition Phase for Thick Ash Deposits (15cm)	For Several Years Immediately After Eruptive Phase Ends	RMEI	43percent of Transition Phase (≤1cm) for AMCG
Steady State Phase for Thin Ash Deposits (≤1cm)	After Transition Phase Ends	RMEI	29percent of Transition Phase (≤1cm) for AMCG
Steady State Phase for Thick Ash Deposits (15cm)	After Transition Phase Ends	RMEI	40percent of Transition Phase (≤1cm) for AMCG

Table 5.2-7 shows that the most conservative BDCFs of the 8 possible sets are the transition phase BDCFs for thin ash deposits (≤1cm) applied to the average member of the critical group. The remaining 7 sets of BDCFs yield doses that are only 29 percent to 79 percent of this set of BDCFs. The transition phase BDCFs for thin ash deposits for the average member of the critical group are the BDCFs that are used within the TSPA-SR. This is a conservative choice for BDCFs because these BDCFs are applicable for the time frame immediately following the eruptive phase of a volcanic eruption. They are used for the full 10,000 years of the model which implies the mass loading of particulates in air remains elevated for the full 10,000 years instead of the expected tens of years. By using the thin ash layer BDCFs, the radionuclides are assumed to be concentrated near the surface. A more realistic assumption for agricultural land in which air mass loading might remain high would be to assume that the radionuclides are plowed into the soil and well mixed, reducing the concentration in the surface layer. Thus, the TSPA-SR use of the thin-layer transition phase eruptive BDCFs is conservative.

Figure 5.2-25 provides insight into the sensitivity of the eruptive dose rate to uncertainty in the values used for the thin-layer transition phase BDCFs by comparing the mean eruptive dose rate to the dose rate calculated with the BDCFs fixed at their 5th and 95th percentile values. The dose rate calculated using the 95th percentile values is approximately a factor of 2 higher than the mean dose rate, reaching a peak of approximately 0.02 mrem/yr.

5.2.9.11 Sensitivity to Incorporation Ratio

Figure 5.2-26 shows a comparison of the probability-weighted mean annual eruptive dose rate, as described in Section 4.2, with the same dose rate calculated using an incorporation ratio of 0.1 and 1.0. As described in Section 3.10.2, the incorporation ratio is used in ASHPLUME 1.4lv (CRWMS M&O 1999 [150744]) to characterize the entrainment of waste particles in the eruption. The base case analysis uses an incorporation ratio of 0.3, which causes waste particles with diameters that are up to 50 percent of the mean diameter of the ash particles to be incorporated in the eruption. The annual eruptive dose rate increases by a negligible factor for an incorporation ratio of 0.1, which increases the diameter of waste particles that are incorporated in the eruption to those that are 80 percent of the ash particle diameter. An incorporation ratio of 1.0, which results in incorporation of particles of up to 10 percent of the ash particle diameter, causes a reduction of less than a factor of 2 in the probability-weighted mean annual dose rate. The relative lack of sensitivity to uncertainty in this parameter suggests that most waste particles are being incorporated in the eruption with the base case value, and that even with the smaller value only the largest waste particles are too small to be incorporated.

5.2.10 Human Intrusion Sensitivity

Section 4.4 describes the TSPA-SR results for the human intrusion scenario. Included in Section 4.4 was a comparison of human intrusion analyses for intrusions occurring at 100 years and 10,000 years after potential repository closure. This section presents the results of an additional one-off sensitivity analysis, in which a selected parameter value (infiltration rate) was fixed and all other parameters were treated as in the human intrusion base case (the case with an intrusion at 100 years). The sensitivity simulation was run for 100 realizations.

For this analysis the infiltration rate was fixed at its 95th percentile value, representative of an extreme unfavorable deviation from expected (base case) performance. Figure 5.2-27 shows a comparison of the mean annual human intrusion dose rate for an intrusion at 100 years (as described in Section 4.4), with the same dose rate calculated using the 95th percentile value for the infiltration rate. The mean annual dose rate increased by a factor of about 5 for the 95th percentile infiltration rate. However, the peak mean annual dose rate for the 95th percentile infiltration rate did not exceed approximately 0.05 mrem/yr in the first 10,000 years after potential repository closure or over the entire 100,000 years.

5.3 ROBUSTNESS ANALYSES

The focus in this section is on the various natural and engineered barriers that make up the potential repository system, and on analyses that examine the robustness of the system by simulating potential repository performance with barriers assumed to be degraded, one at a time or in combination. The objective of such analyses is to evaluate the effectiveness and diversity of the barriers to determine the overall resiliency of the potential repository system to extreme conditions that are unlikely, but within the range of those believed physically possible. These analyses address nominal performance of the system without the occurrence of unlikely disruptive events such as igneous activity.

The robustness analyses were conducted by fixing key parameters affecting the performance of each barrier near the extreme of their uncertainty distributions (either the 5th or 95th percentile, whichever leads to maximizing the dose rate over the time period of interest). By comparing the nominal performance results with the degraded performance results, one can examine the relative contribution of each of the barriers. Although the parameter values used in the degraded barrier analyses are within the range of values considered reasonably possible, the results should not be interpreted as representing the expected behavior of the system. In many of the analyses, several parameters are simultaneously set to unlikely values. The mean nominal performance results (Section 4.1) are the best approximation of the expected behavior of the system in the absence of igneous disruption, but even they contain many conservative assumptions or parameter values, so the base-case dose results are intended to be on the high side of the expected range of behaviors. The degraded barrier analyses are presented only to provide insight into the resilience of the potential repository system to extreme conditions. To ensure a balanced interpretation of the degraded barrier analysis, results are shown paired with comparable analyses using the 5th or 95th percentile values (as appropriate) of the same parameters, which results in more favorable performance. The mean result from the full nominal analysis should be interpreted as the best estimate of future performance, and the degraded- and enhanced-performance analyses should be interpreted as being equally likely (or unlikely) to occur.

The following barriers are considered in the robustness analyses:

- UZ. This barrier represents the function of the UZ above the potential repository in limiting the amount of water that reaches the potential repository. This barrier includes the climatic conditions at Yucca Mountain, the processes at and near the surface that lead to infiltration, and flow through the UZ above the potential repository.

- Seepage into emplacement drifts. This barrier represents the function of the drifts themselves as a capillary barrier that limits the amount of water that enters the drifts.

- Drip shield. The first of the engineered barriers, the drip shield limits the amount of water that reaches the waste package.

- Waste package. The primary engineered barrier, the waste package limits the amount of water that reaches the waste form and limits radionuclide transport out of the EBS.

- CSNF cladding. The Zircaloy cladding is an engineered barrier that is part of the waste form. It limits the amount of water that reaches the CSNF portion of the waste and limits radionuclide transport out of the CSNF waste form. (CSNF is planned to be approximately 90 percent of the mass of waste in the potential repository.)

- Concentration limits. This barrier represents the function of environmental conditions and radionuclide solubility limits in limiting radionuclide transport out of the EBS.

- EBS transport. This barrier represents the function of environmental conditions and diffusion in the drift invert in limiting radionuclide transport out of the EBS.

- UZ transport. This barrier represents the function of the UZ below the potential repository in delaying radionuclide transport to the biosphere.

- SZ. This barrier represents the function of the SZ in delaying radionuclide transport to the biosphere.

Of the nine "barriers" listed above, only two were found to be important in the uncertainty importance analyses described in Section 5.1: the waste package and the SZ. The importance ranking of parameters in the analyses of Section 5.1 is strongly dependent on two factors: the change in variance of dose rate with the variance of the parameter and the change in the dose rate itself with changes in the parameter. Thus a barrier may not appear as important in Section 5.1 if either of these two derivatives is small. For many of the above "barriers", it is the case that the variance derivative is small because the postulated range of uncertainty in the input parameter or model is either small or nonexistent. This is the case for EBS transport, concentration limits (particularly neptunium), and cladding. For others, such as UZ transport and UZ flow, the derivative of dose rate with the key model parameters is small, i.e., changes in dose rate with changes in those model parameters over reasonably expected ranges are small, so those barriers are less important than other barriers such as the waste package.[2] The remaining barriers listed above, i.e., drip shield and seepage, show up as unimportant because their performance is either masked by other processes or they act as redundant barriers. For example, the drip shield is redundant to the waste package and generally has a shorter lifetime, so its effect is masked by waste package performance. Similarly, the effect of seepage is masked by diffusive transport, which is quite high compared to advection in the first 50,000 years or so.

Another important point is that the uncertainty importance analyses in Section 5.1 are based on linear regression, whereas the TSPA-SR model is often nonlinear in its response. Thus, the overall fit of the regression model is not high enough to evaluate the importance of some of the lesser tier parameters, such as those related to UZ performance.

This section on robustness is another method to evaluate the importance of the various barriers, in order to help bolster or negate the case made in Section 5.1. Robustness of the system with respect to a given barrier will be demonstrated by "small" or negligible changes in the dose rate. Conversely, large changes in dose rate for a given change in a barrier parameter tend to indicate that the system is not as robust with respect to that barrier, i.e., adequate performance of that barrier is more critical to overall system performance than other less important barriers. As will be confirmed below with respect to each of the nine listed barriers, the robustness analyses of this section tend to confirm the results of Section 5.1. However, since all of the robustness analyses in this section necessarily stay within the base-case uncertainty ranges of the individual

[2] There is some disagreement on this point and one could argue that because of the nature of the underlying conceptual model used to model the UZ, it is not possible to show a great impact on dose rate. Perhaps an alternative conceptual model, e.g., one with less conservative assumptions, would reveal a greater influence. Also, as described in the next paragraph, linear regression may not be the most appropriate method for analyzing a highly nonlinear model.

parameters, they cannot elevate in importance any parameter or barrier which has a very restricted range of uncertainty.

The robustness analyses in this section were performed using probabilistic TSPA simulations with 100 realizations. One hundred realizations were used rather than the 300 used for the base-case simulations (see Sections 4.1 and 4.2) because 100 realizations are sufficient to see the relative effects when comparison is made to the base case (see Section 4.1.4).

5.3.1 Unsaturated Zone Flow

5.3.1.1 Degradation of Unsaturated Zone Flow

For purposes of these robustness analyses, UZ flow is degraded by considering the high-infiltration case and it is enhanced by considering the low-infiltration case. Changing the amount of infiltration affects the whole UZ system: the amount of seepage that enters emplacement drifts, the thermal hydrologic environment in the drifts, and the radionuclide transport time through the UZ. The degraded and enhanced UZ flow analyses are the same as the one-off analyses of infiltration discussed in Section 5.2.1.1. There, it was shown that there is only a small increase in doses for the degraded UZ flow case, but a significant decrease in doses for the enhanced UZ flow case.

5.3.1.2 Degradation of Seepage into Drifts

Degraded seepage conditions were defined as

- The seepage-uncertainty factor at its 95th percentile value
- The seepage flow-focusing factor at its 95th percentile value.

while enhanced seepage was defined as

- The seepage-uncertainty factor at its 5th percentile value
- The seepage flow-focusing factor at its 5th percentile value.

The seepage uncertainty factor is a uniform random variable that is used to correlate the sampling of seepage fraction, mean seep flow rate, and the standard deviation of seep flow rate from their respective distributions (see Section 3.2.4.3 for discussion of these parameters). When the seepage-uncertainty factor is at its 95th percentile value, all three of those seepage parameters are at their 95th percentile values, and similarly for the 5th percentile values. Thus, the amount of seepage flow and the number of waste packages affected are both high when the seepage-uncertainty factor is high, and both are low when it is low.

The effect of the flow-focusing factor is less straightforward. As discussed in Section 3.2.4.3, a higher flow-focusing factor causes higher seep flow rates for the waste packages that have seepage, but at the same time reduces the number of waste packages affected by seeps. The combined effect, as discussed in Section 5.2.1.2, is an increase in the total amount of seepage and an increase in radionuclide releases. Similarly, seepage and radionuclide releases decrease when the flow-focusing factor is decreased. Thus, an increased flow-focusing factor was included in the degraded seepage case and a decreased flow-focusing factor was included in the enhanced seepage case.

Figure 5.3-1 shows a comparison of the mean annual nominal dose rate to the mean dose rates calculated under degraded and enhanced seepage conditions. The impact of degraded and enhanced seepage conditions starts to be seen at about 40,000 years. This is due to the waste packages not having failed "patches" (i.e., advection release pathways) until about 36,000 years, plus a travel time of approximately 4,000 years from the EBS to the critical-group location. Higher percolation fluxes due to the 95th percentile flow-focusing factor, along with higher seepage flows from the 95th percentile seepage-uncertainty factor result in higher mean dose rates in comparison to the nominal case. Conversely, the low percolation fluxes associated with the 5th percentile flow-focusing factor, along with lower seepage flows due to the 5th percentile seepage-uncertainty factor result in lower doses rates in comparison to the nominal case. The calculated increases and decreases in dose are only moderate, amounting to increases or decreases by a factor of at most 5 from the base case, and less than that most of the time.

5.3.1.3 Degradation of Unsaturated Zone Flow and Seepage

This sensitivity case is a combination of degraded UZ flow (Section 5.3.1.1) and degraded seepage (Section 5.3.1.2). That is, the seepage parameters are set to the pessimistic end of their uncertainty ranges and in addition infiltration is set to its high case. The corresponding enhanced UZ flow and seepage case has seepage parameters set to the optimistic end of their uncertainty ranges and infiltration set to its low case.

The results of these cases are presented in Figure 5.3-2. Shown in the plot are the mean nominal-scenario dose curves for the degraded and enhanced UZ flow and seepage cases, along with the mean dose curve for the nominal-scenario base case, for comparison. The results are seen to be a combination of the results of the individual cases (see Figures 5.2-1 and 5.3-1). The largest effect comes from the reduced infiltration, which greatly increases the transport time through the UZ (see Figure 3.7-10). The mean doses from the enhanced UZ flow and seepage case are significantly lower than the base case. However, the combination of high infiltration and degraded seepage still results in doses that are only moderately higher than the base case: increases by a factor of up to about 5 or so.

5.3.2 Engineered Barrier System Environments

No barrier degradation analyses were performed for the EBS environments alone; however, some aspects of the environments are modified along with other parameters for the analyses described in Sections 5.3.4.2 and 5.3.5.

5.3.3 Waste Package and Drip Shield Degradation

5.3.3.1 Degradation of Drip Shield

In the current EBS transport model, the titanium drip shield must be failed before any advective (flowing) water can contact the waste packages and carry away radionuclides from the failed waste packages. Analyses were conducted to evaluate the impact of drip shield performance on potential repository performance. The analyses were conducted by fixing the drip shield degradation parameters at the 95th and 5th percentile values of their respective uncertainty distributions. The cases are referred to as the degraded drip shield case (95th percentile case) and the enhanced drip shield case (5th percentile case). Only the drip shield general corrosion is considered because it is the only active degradation mode in the base-case analysis (see Sections 3.4.1.3 and 3.4.1.5 for an explanation of why other modes are unimportant). The degraded case uses the 5th percentile uncertainty-variability partitioning ratio from the parent distribution of the titanium general corrosion rate (i.e., the spread in the parent distribution from the experimental measurements is considered to represent mostly variability, rather than uncertainty). It also sets the median general corrosion rate to the 95th percentile of the resulting uncertainty variance (i.e., the variance in shield-to-shield corrosion rates that results from setting the uncertainty-variability partitioning ratio to its 5th percentile). The enhanced case uses the 95th percentile uncertainty-variability partitioning ratio and the median general corrosion rate at the 5th percentile of the resulting uncertainty variance.

The analysis results for the predicted mean dose rate profile and mean drip shield failure profile are shown in Figures 5.3-3 and 5.3-4 respectively. Although the analyses show that the drip shield performance is affected significantly (Figure 5.3-4), there is almost no effect on the predicted mean dose rate (Figure 5.3-3). This is due primarily to the fact that waste package degradation is independent of drip shield performance in the WAPDEG model, (i.e., none of the individual corrosion models or parameters is affected by drips). As discussed above, the intact drip shields prevent dripping water from directly contacting the underlying waste packages, and this should affect the EBS transport process. However, this benefit appears not to be significant to the mean dose rate (Figure 5.3-3).

5.3.3.2 Degradation of Waste Package

This section analyzes the sensitivity of the potential repository and waste package performance to several major waste package degradation process parameters. Analyses were conducted by fixing the degradation parameters at the 95th and 5th percentile values of their respective uncertainty distributions. The cases are referred to as the degraded waste package case (95th percentile case) and the enhanced waste package case (5th percentile case). The degradation parameters considered are:

- Residual hoop stress state and stress intensity factor at the closure-lid welds
- Number of manufacturing defects at the closure-lid welds per waste package
- Alloy-22 general corrosion rate
- Enhancement factor to Alloy-22 general corrosion due to MIC
- Enhancement factor to Alloy-22 general corrosion due to aging and phase stability.

Note that individual variations of several of these parameters are presented in Section 5.2.3. In this section all of them are varied together to represent significantly degraded waste package performance. The degraded case uses:

- The 5th percentile uncertainty-variability partitioning ratio from the parent distribution of the Alloy-22 general corrosion rate (i.e., the spread in the parent distribution from the experimental measurements is considered to represent mostly variability, rather than uncertainty)

- The median general corrosion rate at the 95th percentile of the resulting uncertainty variance (i.e., the variance in package-to-package corrosion rates that results from setting the uncertainty-variability partitioning ratio to its 5th percentile)

- The uncertainty indices for the residual hoop stress and stress intensity factor in both the outer and middle lids at their 95th percentile values, implying earlier SCC failure than the base case

- The manufacturing defect parameters at their 95th percentile value, which maximizes the defects

- The MIC enhancement factor at its 95th percentile value

- The aging and phase stability enhancement factor at its 95th percentile value.

The enhanced case uses:

- The 95th percentile uncertainty-variability partitioning ratio

- The median general corrosion rate at the 5th percentile of the resulting uncertainty variance

- The uncertainty indices for the residual hoop stress and stress intensity factor in both the outer and middle lids at their 5th percentile values, implying later SCC failure than the base case

- The manufacturing defect parameters at their 5th percentile value, which minimizes the defects

- The MIC enhancement factor at its 5th percentile value

- The aging and phase stability enhancement factor at its 5th percentile value.

See Section 3.4 for the discussions on the enhancement factors due to microbiologically influenced corrosion and aging and phase stability. Also, it should be noted that in each of the above two cases there are a total of 9 parameters that are fixed, so the probability of this case ever being sampled is extremely small, on the order of 0.05^9 or about 10^{-12}.

The analysis results for the predicted mean dose rate profile and mean waste package failure profile are shown in figures 5.3-5 and 5.3-6, respectively. The enhanced case yielded no waste package failure, thus no dose, during the first 100,000 years. On the other hand, by fixing the major degradation parameters at their "degraded" values, the waste package degradation rate increases significantly, and the failure rates are significantly higher (Figure 5.3-6). The first failure time of the mean failure profile is about 7,000 years, compared to about 12,000 years for the base case. For the degraded case, there is 50 percent probability that 1 percent of waste packages fail at about 10,000 years and 10 percent of waste packages fail at about 12,000 years. For the base case it is about 25,000 years for the 1 percent failure and about 50,000 years for the

10 percent failure. Accordingly, the predicted mean dose starts earlier (about 8,200 years versus about 15,000 for the base case), and the predicted mean dose rates are much higher. These results are consistent with the individual sensitivity analyses presented in Section 5.2.3.

5.3.4 Waste Form Degradation and Mobilization

5.3.4.1 Degradation of CSNF Cladding

For the two most important radionuclides to nominal-scenario dose, ^{99}Tc and ^{237}Np, the CSNF represents over 86 and 95 percent of the total inventory, respectively. Hence, the CSNF cladding can directly influence the dose by reducing the release rate of these radionuclides.

The cladding model has five parameters that were sampled for TSPA-SR: (1) the number of rods initially perforated in a CSNF waste package (f_{rod}) (Initial_Rod_Failures), (2) the fraction of cladding perforated because of creep rupture and SCC (Creep_Used), (3) the uncertainty in localized corrosion rate (LC_uncert), (4) the uncertainty of the CSNF degradation rate (Uncert_a0), and (5) the uncertainty in the unzipping velocity of the cladding (Unzip_uncert) (Table 5.3-1). An estimate of the uncertainty that the cladding model causes in the dose was evaluated by setting all sampled parameters except the fraction of cladding perforated by creep rupture and SCC at the 5 percent and 95 percent values and observing the change in the mean dose (Figure 5.3-7). The mean dose only increases slightly (about a factor of 1.5) when the four parameter values are set at the 95th percentiles of their distributions. When the four parameters are set at their 5th percentiles, the mean dose decreases by about a factor of 4 in the first 100,000 years.

Table 5.3-1. Sampled Parameters in CSNF Cladding Degradation Model

Parameter	Description	Parameter Distribution
Initial_Rod_Failures	Percentage of cladding with initial perforation	Triangular (0.0155, 0.0948, 1.29)
Creep_Used	Fraction of cladding perforated due to creep rupture and SCC	Triangular (limits are a function of maximum temperature calculated in waste package for each simulation)
LC_uncert	Uncertainty in the localized corrosion rate	Loguniform (0.1, 10)
Uncert_a0	Uncertainty in the CSNF degradation rate (10^x)	Uniform (–1, 1)
Unzip_uncert	Uncertainty in unzipping velocity	Triangular (1, 40, 240)

Until several tens of thousands of years after the waste package begins to fail, the only cladding perforated is CSNF cladding that arrives at the site perforated or perforates because of creep rupture during the first few hundred years. For example, the mean perforation fractions are respectively 0.0045 and 0.0765 for initial perforation and creep rupture for the 20 to 60 mm/yr infiltration bin. Only after 50,000 years does the perforated fraction of cladding change due to localized corrosion in those waste packages that have seepage (Figure 4.1-10). As described in Section 3.5, the localized corrosion is a direct function of the seepage volume into the waste package. The sometimes-dripping case has the greater mean seepage volume; thus, the sometimes-dripping case has the most localized corrosion and greater perforation.

Overall, parameters related to the waste form degradation model or the cladding component, in particular, are not important to determining uncertainty in the dose in the first 100,000 years (see Section 5.1). However, between 100,000 years and 1,000,000 years, uncertainty in two parameters of the cladding component show up as important to determining uncertainty in the dose. During this period, more cladding begins perforating from localized corrosion. Of the five parameters varied in the cladding component, the order-of-magnitude uncertainty in the CSNF degradation rate was found to be the most influential (Figure 5.1-18). In addition, the two-order-of-magnitude uncertainty in the unzipping rate was found to be important at some times.

5.3.4.2 Degradation of Concentration Limits

The dose rate out to 100,000 years is relatively insensitive to the concentrations of dissolved and colloidal radionuclides in the waste package and invert. That conclusion is illustrated in Figure 5.3-8. For the degraded concentration case, parameters affecting concentrations were set to values at the pessimistic end of their uncertainty ranges. Radionuclide concentrations were set to their 95th percentile values, colloid stability was maximized, and the 95th percentile values were used for sorption coefficients of radionuclides onto colloids. In addition, the waste package chemistry was used in the invert, because the chemistry in the waste package generally favors higher concentrations in the TSPA model. The lack of importance of solubility is explained as follows.

The two most important radionuclides to dose in the first 100,000 years are ^{99}Tc and ^{237}Np. In the TSPA model, the solubility of technetium is set to a constant bounding value of 1 M; thus, although ^{99}Tc is still the most important radionuclide for determining the dose in the first ~30,000 years after waste package failure (see Section 4.1), uncertainty in technetium solubility does not influence the uncertainty in the total dose because it is a constant.

^{237}Npis the most important radionuclide for determining the dose after ~30,000 years, but its solubility does not influence the uncertainty in the dose either; however, the reason for its low importance is different. In the TSPA model, the solubility of neptunium is a function of the pH. Inside the waste packages, the pH varies substantially; consequently, the solubility of neptunium also varies (Figure 3.5-21). Yet, outside the waste package in the invert, the pH does not vary much. The TSPA model reevaluates the neptunium solubility at the invert using the invert pH (any excess ^{237}Np diffuses out of the invert or is held in the invert until the concentration drops at later times). A situation similar to ^{237}Np also occurs for uranium and americium since they are a function of pH as well. However, they are lesser contributors to the total dose for nominal performance.

Because the invert pH does not vary much, neither does the neptunium solubility in the invert. Thus, the calculated variability of the neptunium solubility is not large enough to substantially influence the variability in the dose (Figure 5.3-8); therefore, neptunium solubility does not show up as an important parameter in sensitivity analysis for TSPA-SR. As discussed in Section 5.2.4.2, the neptunium solubility used in TSPA-SR is probably somewhat conservative and does not account for the formation of secondary mineral phases. Section 5.2.4.2 also presents a sensitivity analysis showing how potentially important neptunium solubility could be if it had a wide range of variation.

In the first 100,000 years, all three waste forms (i.e., HLW glass, DSNF waste matrix, and CSNF waste matrix exposed from unzipping of initially perforated cladding) liberate radionuclides faster than radionuclides are released from the waste package through either diffusion or advection. Thus, large amounts of various radionuclides are retained in the package prior to release (see, e.g., Figure 4.1-11). Hence, the waste-matrix degradation rate is unimportant to determining the dose and its uncertainty. As just discussed, the release is not regulated by the solubility of the radionuclides to a great extent, either. Rather, since most of the release of radionuclides is from diffusive rather than advective release (Figure 4.1-13), it is limited flux rate of material through the limited available surface area of the package that controls the release. Consequently, even the highly soluble ^{99}Tc is retained in the waste package for long periods of time as described in Section 4.1.

5.3.5 Engineered Barrier System Transport

This section analyzes the sensitivity of the potential repository to several important EBS transport and EBS environment parameters. The degraded EBS transport case is a combination of the degraded concentration limits case (Section 5.3.4.2) and the high-diffusion case (Section 5.2.5), while the enhanced EBS transport case is a combination of the enhanced concentration limits case (Section 5.3.4.2) and the low-diffusion case (Section 5.2.5). As defined in those earlier sections, "degraded concentration limits case" means high solubilities and "enhanced concentration limits case" means low solubilities.

The combined effects of EBS transport and related chemical environments on the early-arrival dose rate (time to a dose rate of 10^{-3} mrem/yr) are differences of several thousand years. Furthermore, the peak dose rate at 100,000 years is moderately sensitive to the diffusion model and to the effects of chemical environments on the concentrations of dissolved and colloidal radionuclides in the EBS. This is illustrated in Figure 5.3-9 by the factor of 5 increase in peak dose rate for the degraded EBS transport case.

5.3.6 Unsaturated Zone Transport

The following three subsections present robustness analyses related to UZ transport. Section 5.3.6.1 presents a degraded UZ transport analysis in which several UZ transport parameters are set at pessimistic values (i.e., values that put radionuclide transport time in the fast end of its uncertainty range). Section 5.3.6.2 combines the degraded UZ transport parameters with degraded UZ flow (i.e., high infiltration). Lastly, Section 5.3.6.3 combines the degraded UZ transport and UZ flow with degradation of seepage into drifts as well (seepage parameters set at pessimistic values). Each of these sections also presents the converse: the result if the same parameters are set to optimistic values. In this way, the effect on the calculated dose of a range of behaviors, from pessimistic to optimistic, can be evaluated.

5.3.6.1 Degradation of Unsaturated Zone Transport

To examine the effects of degraded UZ transport, the following parameters were set to values at the pessimistic end of their uncertainty ranges:

- All radionuclide sorption coefficients (K_ds) were set to their 5th percentile values (low sorption implies faster transport).
- All colloid partitioning factors (K_cs) were set to their 95th percentile values (high K_c implies faster transport of the reversible colloids).
- All diffusion coefficients were set to their 5th percentile values (low matrix diffusion implies faster transport).
- All fracture apertures were set to their 95th percentile values (large fracture apertures lead to less matrix diffusion, which implies faster transport).

In the corresponding enhanced UZ transport case, the above parameters were set to the optimistic end of their uncertainty ranges (that is, the 5th and 95th designations above were switched: K_ds at 95th percentile values, etc.). All other parameters were given the same (sampled) values as in the nominal-scenario base case.

The results of these cases are presented in Figure 5.3-10. Shown in the plot are the mean dose curves for the degraded and enhanced UZ transport cases, along with the mean dose curve for the nominal-scenario base case, for comparison. The effect of the degraded or enhanced UZ transport is small to moderate except for a large pulse in the degraded case between 20,000 and 30,000 years. That pulse is a result of a large release of ^{243}Am when waste packages start failing, which is then able to transport through the UZ relatively quickly and reach the biosphere before it decays (^{243}Am has a half-life of only about 7,000 years, so the longer transport times in the enhanced and base cases are enough to reduce its dose effect substantially). Comparison of the degraded case in Figure 5.3-10 with Figure 5.2-14 indicates that among the four degraded parameter distributions considered (i.e., apertures, sorption coefficients, K_cs, and matrix diffusion coefficient), matrix diffusion is apparently the most important parameter contributing to the attenuation of dose.

5.3.6.2 Degradation of Unsaturated Zone Flow and Transport

This sensitivity case is a combination of degraded UZ transport (Section 5.3.6.1) and degraded UZ flow (Section 5.3.1.1), i.e., transport parameters are set to the pessimistic end of their uncertainty ranges and in addition infiltration is set to its high case. As with the other degradation cases, a corresponding enhanced case is presented as well.

The results of these cases are presented in Figure 5.3-11. Shown in the plot are the mean nominal-scenario dose curves for the degraded and enhanced UZ flow and transport cases, along with the mean dose curve for the nominal-scenario base case, for comparison. The biggest change from Figure 5.3-10 (degraded UZ transport) is in the enhanced case: the low infiltration significantly increases the transport time through the UZ (see Figure 3.7-10). The mean doses from the enhanced UZ flow and transport case are significantly lower than the base case. The combination of high infiltration and degraded UZ transport results in a large pulse at 20,000 to 30,000 years, as in Figure 5.3-10 and in addition gives doses about a factor of five higher than the base case at late times.

5.3.6.3 Degradation of Unsaturated Zone Flow and Transport and Seepage

This sensitivity case is a combination of degraded UZ transport (Section 5.3.6.1), degraded UZ flow (Section 5.3.1.1), and degraded seepage (Section 5.3.1.2), i.e., transport and seepage parameters are set to the pessimistic end of their uncertainty ranges and in addition infiltration is set to its high case. This case represents the degradation of all aspects of the UZ model. A corresponding enhanced case is presented as well.

The results of these cases are presented in Figure 5.3-12. Shown in the plot are the mean dose curves for the degraded and enhanced UZ flow, transport, and seepage cases, along with the mean dose curve for the nominal-scenario base case, for comparison. The curve for the enhanced case is essentially identical to the corresponding curve in Figure 5.3-11 (degraded UZ flow and transport), indicating that the enhancement of seepage had no additional effect beyond the enhancement of UZ flow and transport. The doses in the degraded case are somewhat higher than the corresponding doses for degraded UZ flow and transport: more than a factor of ten higher than the base case at some times.

5.3.7 Saturated Zone Flow and Transport

5.3.7.1 Degradation of Saturated Zone Flow and Transport

To evaluate the robustness of the SZ as a barrier to radionuclide release from a potential repository at Yucca Mountain, the SZ flow and transport model was parameterized for two cases: one case would give degraded behavior when compared with the base case; the other would give enhanced behavior.

To achieve degraded behavior, the 5th percentile value was taken from distributions of parameters known to positively affect radionuclide travel time, and the 95th percentile value was taken from distributions of parameters known to negatively affect radionuclide travel time. For example, sorption is known to positively affect travel time, because an increase sorption causes an increase in travel time, so, for the degraded SZ, the 5th percentile value for sorption coefficients (K_d) used. Similarly, groundwater flux is known to negatively affect travel time, because an increase groundwater flux causes a decrease in travel time; thus, for the degraded SZ, the 95th percentile value for groundwater flux was used. Conversely, to achieve enhanced behavior, the 95th percentile value was taken from distributions of parameters known to positively affect radionuclide travel time, and the 5th percentile value was taken from distributions of parameters known to negatively affect radionuclide travel time.

Figure 5.3-13 presents the mean nominal-scenario dose calculated from 100 realizations for the base case, the base case with a degraded SZ, and the base case with an enhanced SZ. The difference in dose between the degraded and the enhanced cases is between one and two orders of magnitude (between a factor of 10 and a factor of 100). Two other features are immediately noticeable in the plot: the degraded SZ is very similar to the base case, and the performance from the degraded and the enhanced cases is diverging at late time.

The performance of the degraded SZ is similar to the base case because the realizations that contribute to the largest doses, the doses that have the largest influence on the mean, are realizations that sample an SZ with characteristics similar to the degraded SZ. For example, the realizations with the largest doses in the base case are those using an SZ with a large groundwater flux. The degraded SZ has a high groundwater flux. Groundwater flux is singled out here because it is a parameter that is identified in the regression analyses as influencing dose (Section 5.1). The correspondence of the means of the degraded case and the base case implies that the SZ results are skewed by the least favorable realizations.

The dose from the enhanced SZ is leveling off at 100,000 years, while the dose from the degraded SZ and the base case is continuing to increase. The implication is that the performance of the two cases tends to diverge farther over longer periods of time. The reason for this divergence is that the enhanced SZ effectively reduces the impact of adsorbing radionuclides, such as neptunium, as well as radionuclides that undergo colloid-facilitated transport, such as plutonium, but not the unreactive radionuclides, such as technetium. Technetium is beginning to wane by 100,000 years, explaining the leveling of the enhanced SZ curve.

5.3.8 Biosphere

The biosphere is the endpoint for the TSPA simulation of radionuclide release and transport, and as such is not treated as a barrier to radionuclide contamination. An evaluation of the sensitivity of the TSPA dose calculation to uncertainty in modeling the biosphere is presented in Section 5.2.8. In addition, an evaluation of uncertainty internal to the biosphere model (for groundwater releases only), separate from the TSPA model, is presented in Section 3.9.2.5.

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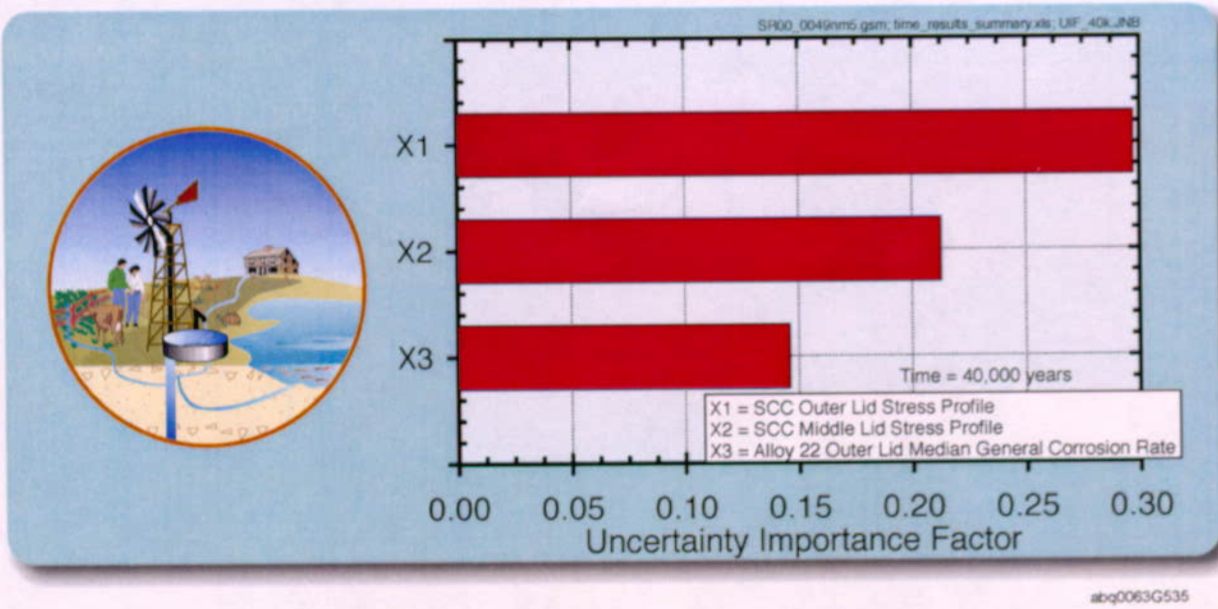

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Figure 5.1-1. Bar Chart Showing Uncertainty Importance Factors for the Most Important Variables at 40,000 Years for Total Dose, Nominal Scenario

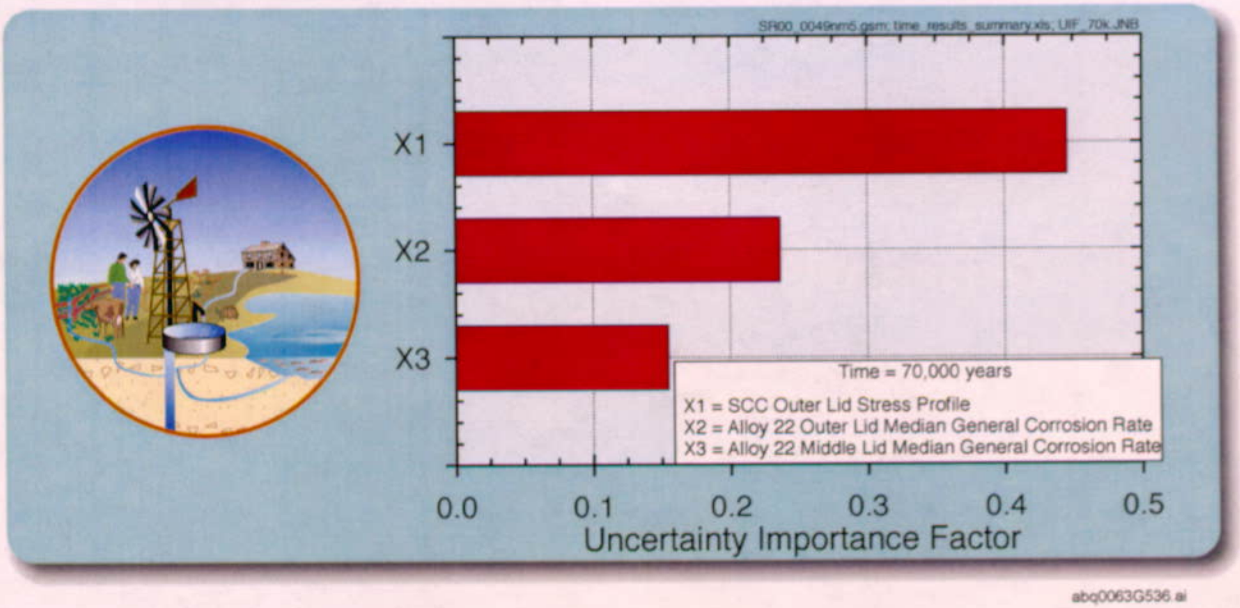

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Figure 5.1-2. Bar Chart Showing Uncertainty Importance Factors for the Most Important Variables at 70,000 Years for Total Dose, Nominal Scenario

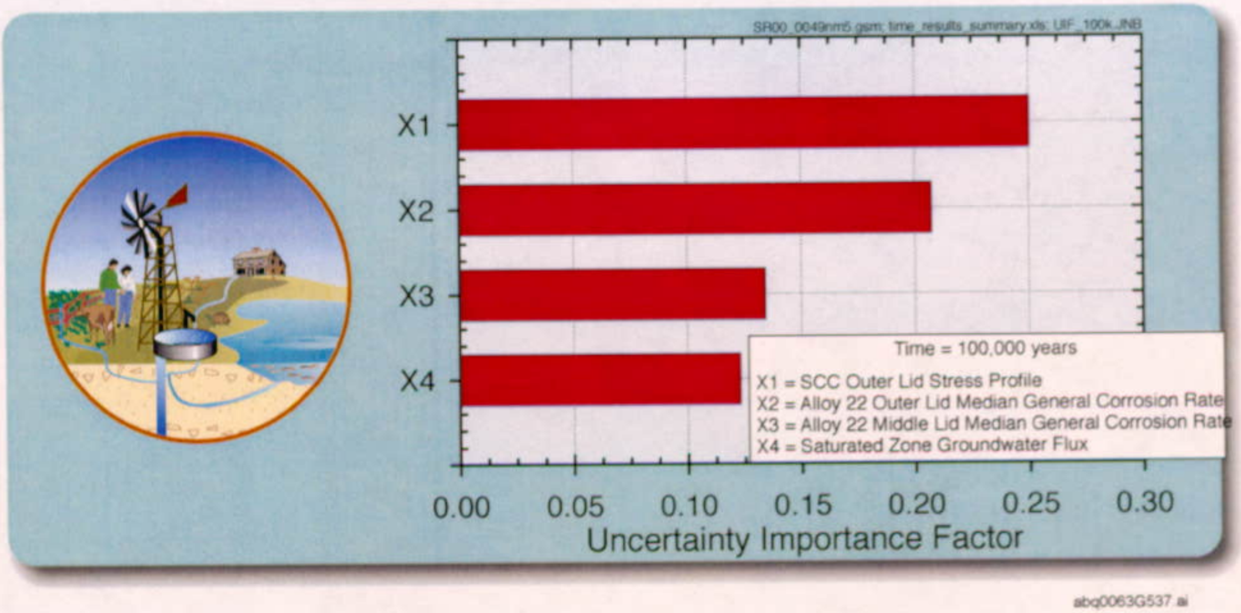

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Figure 5.1-3. Bar Chart Showing Uncertainty Importance Factors for the Most Important Variables at 100,000 Years for Total Dose, Nominal Scenario

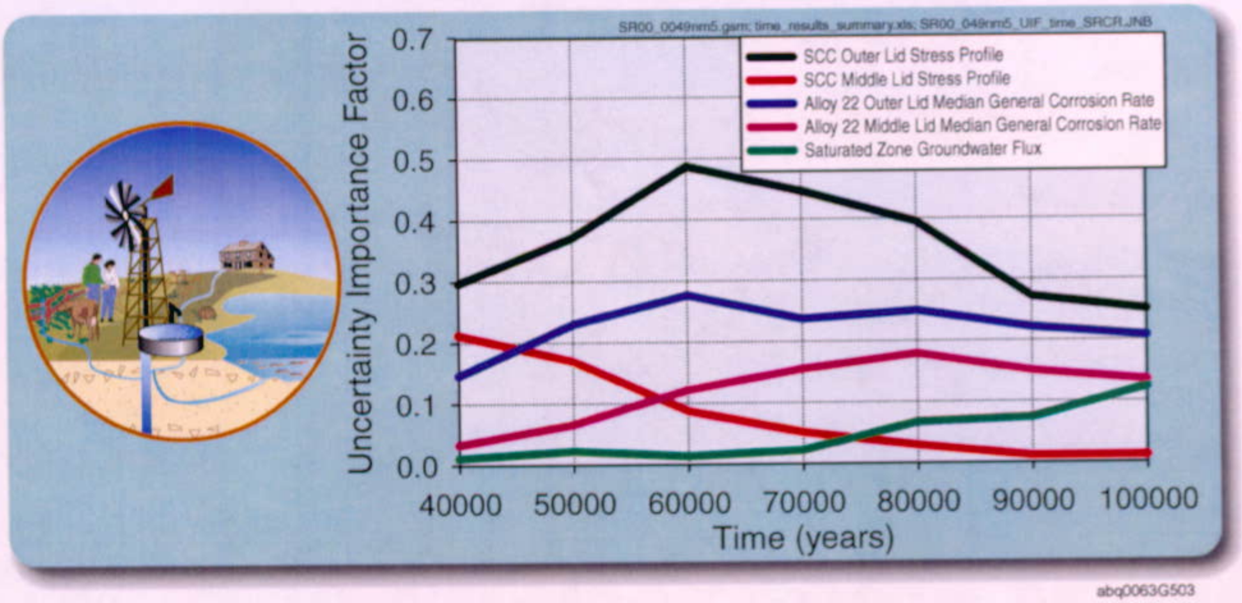

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Figure 5.1-4. Uncertainty Importance Factors at Multiple Time Slices for Total Dose, Nominal Scenario

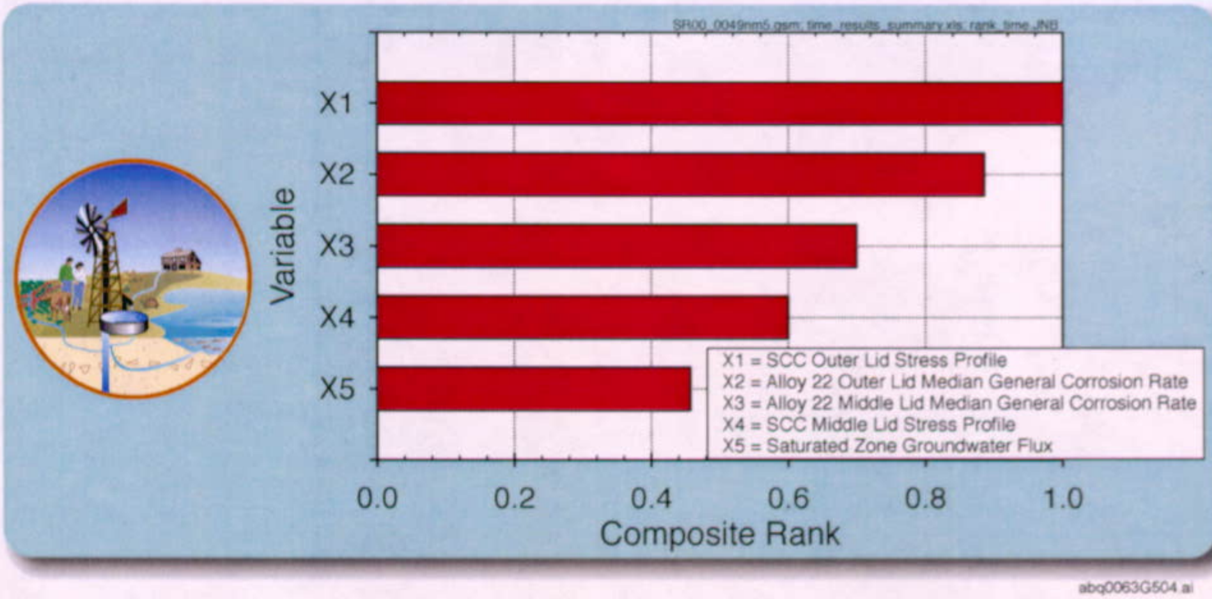

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Figure 5.1-5. Bar Chart Showing Time-Averaged Composite Uncertainty Importance Ranking for Total Dose, Nominal Scenario

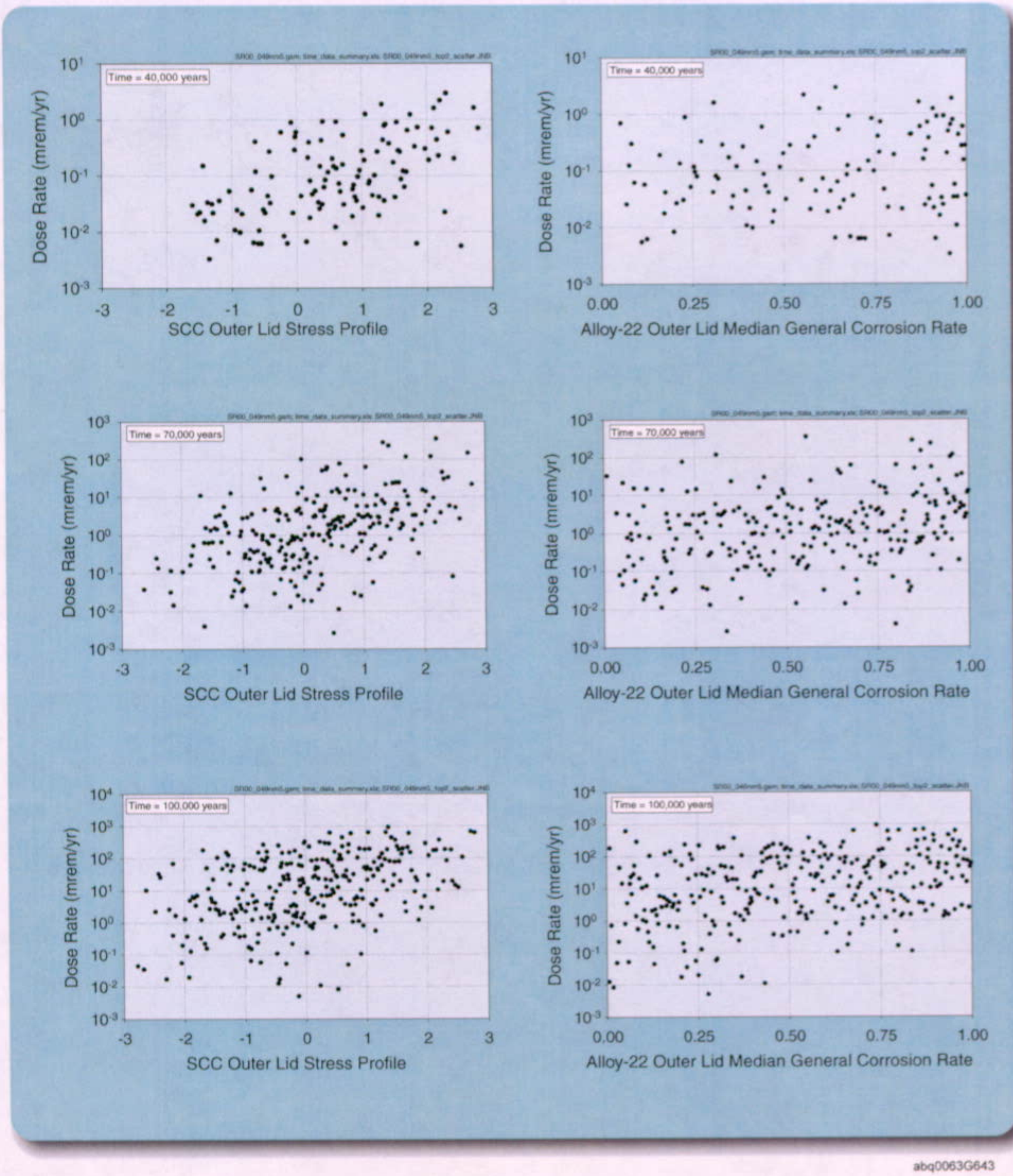

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Figure 5.1-6. Scatter Plots of Total Dose and the Two Most Important Uncertain Variables at Multiple Time Slices, Nominal Scenario

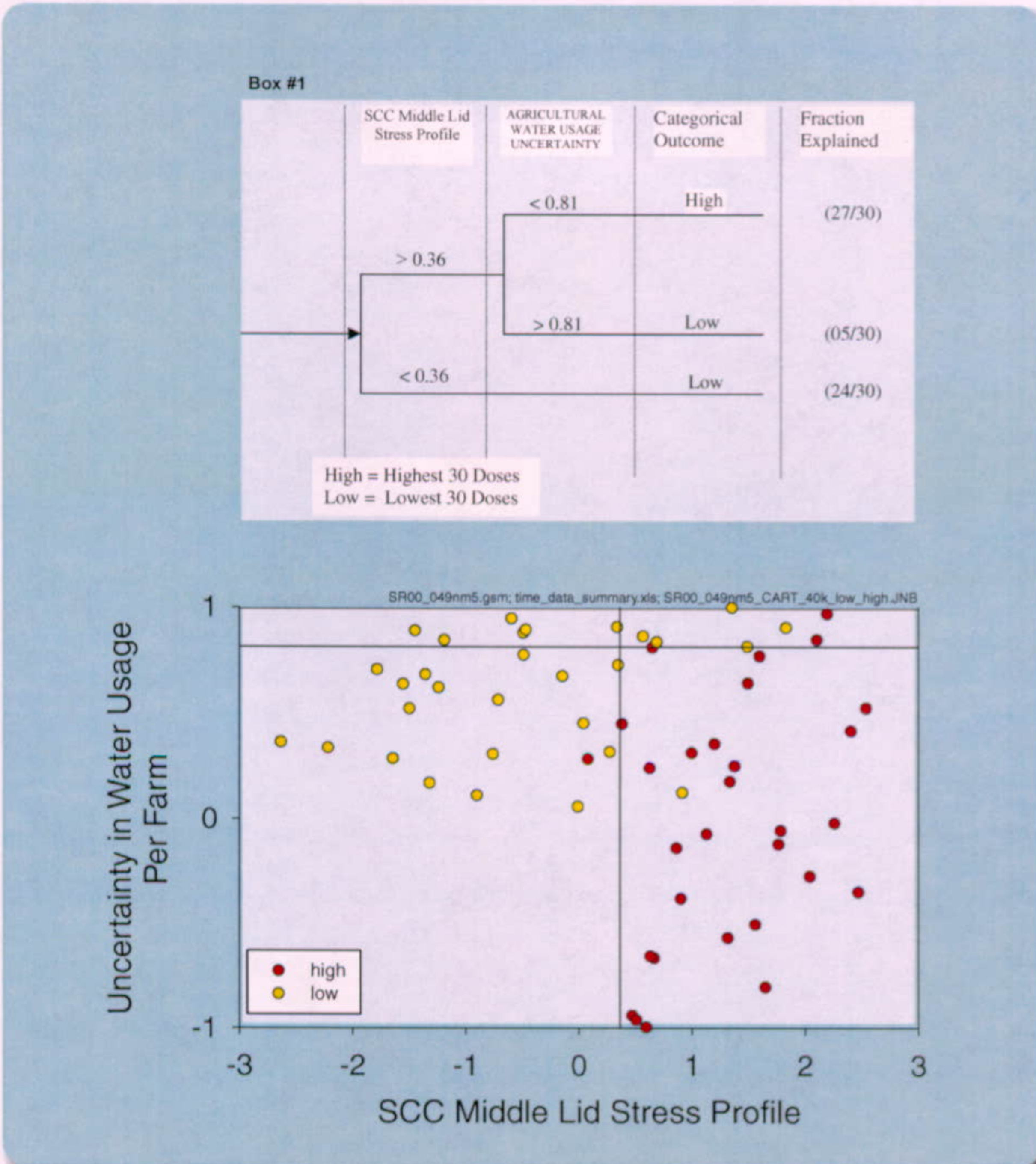

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Figure 5.1-7. Decision Tree Summarizing Classification Tree Analysis (Top) and Partition Plot Showing Clusters of Low-Dose (10th-Percentile and Lower) and High-Dose (90th-Percentile and Higher) Outcomes (Bottom) for the Two Most Important Variables at 40,000 Years, Nominal Scenario

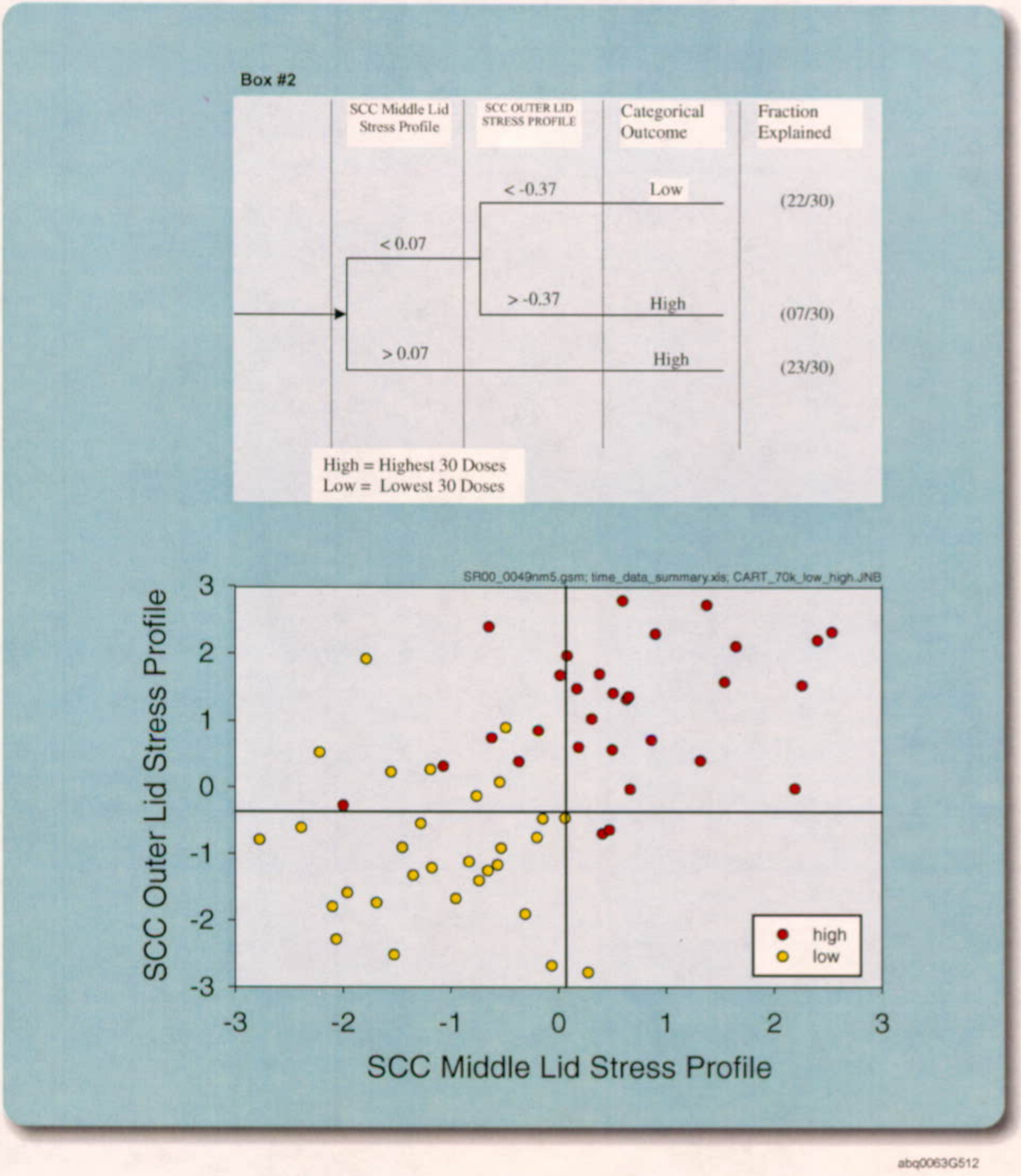

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Figure 5.1-8. Decision Tree Summarizing Classification Tree Analysis (Top) and Partition Plot Showing Clusters of Low-Dose (10th-Percentile and Lower) and High-Dose (90th-Percentile and Higher) Outcomes (Bottom) for the Two Most Important Variables at 70,000 Years, Nominal Scenario

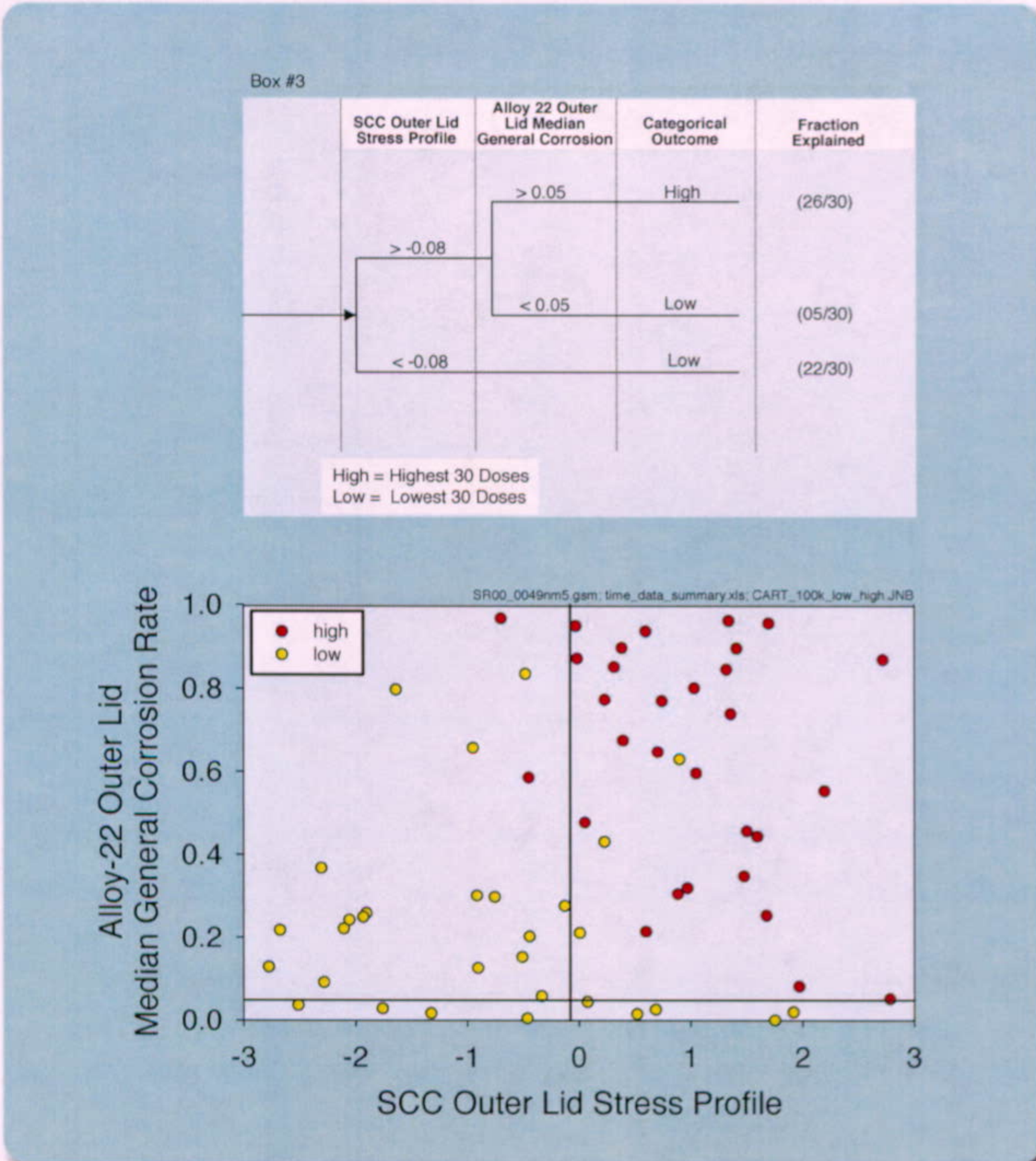

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Figure 5.1-9. Decision Tree Summarizing Classification Tree Analysis (Top) and Partition Plot Showing Clusters of Low-Dose (10th-Percentile and Lower) and High-Dose (90th-Percentile and Higher) Outcomes (Bottom) for the Two Most Important Variables at 100,000 Years, Nominal Scenario

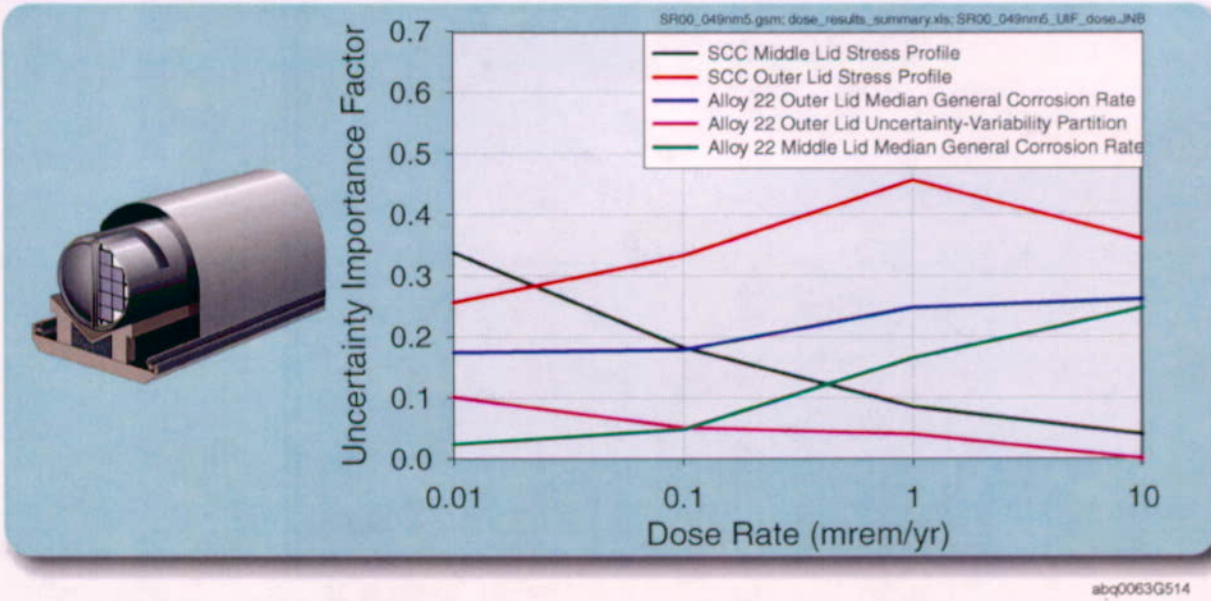

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Figure 5.1-10. Uncertainty Importance Factors at Multiple Dose Values for Time to Reach a Specified Dose, Nominal Scenario

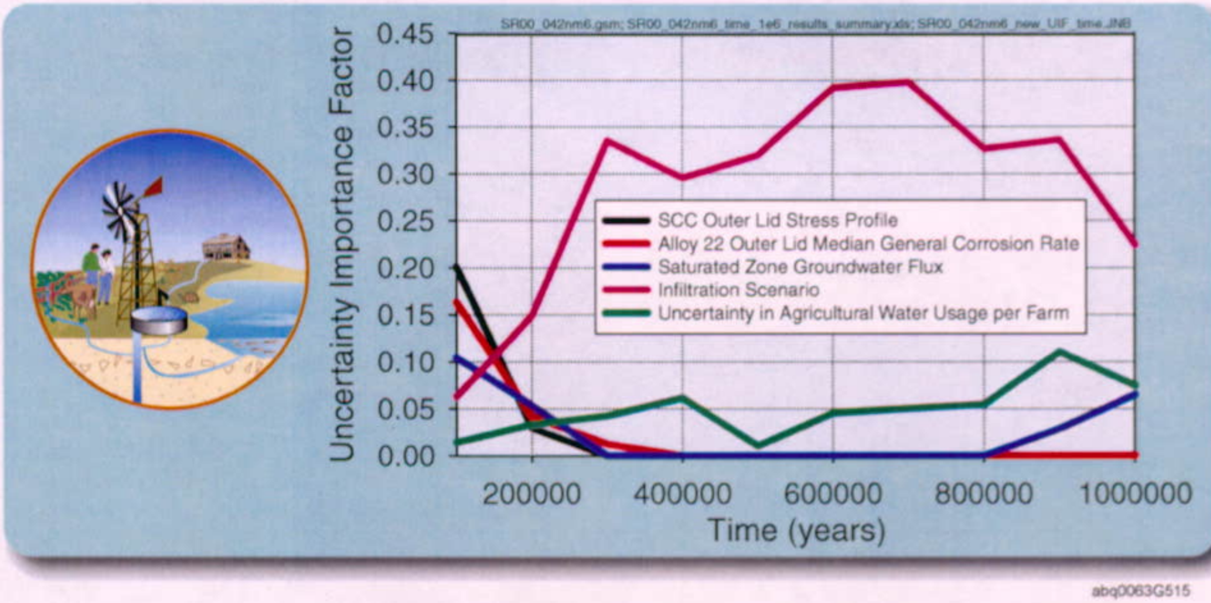

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Figure 5.1-11. Uncertainty Importance Factors at Multiple Time Slices Between 100,000 and 1,000,000 Years for Total Dose, Nominal Scenario

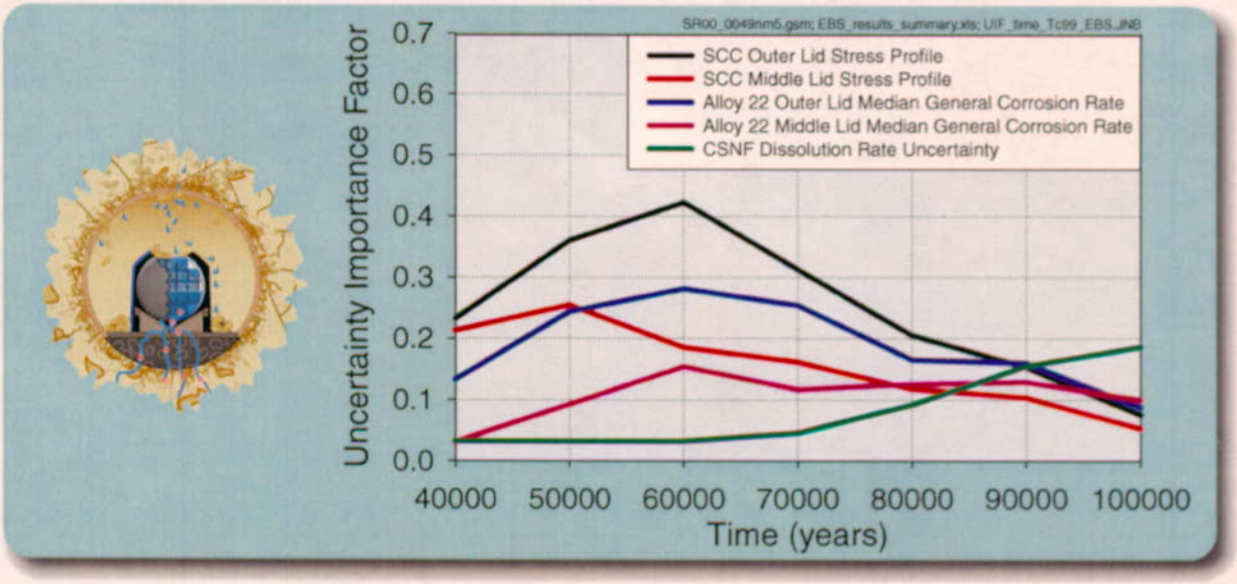

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Figure 5.1-12. Uncertainty Importance Factors at Multiple Time Slices for Mass Release of ^{99}Tc from Engineered Barrier System, Nominal Scenario

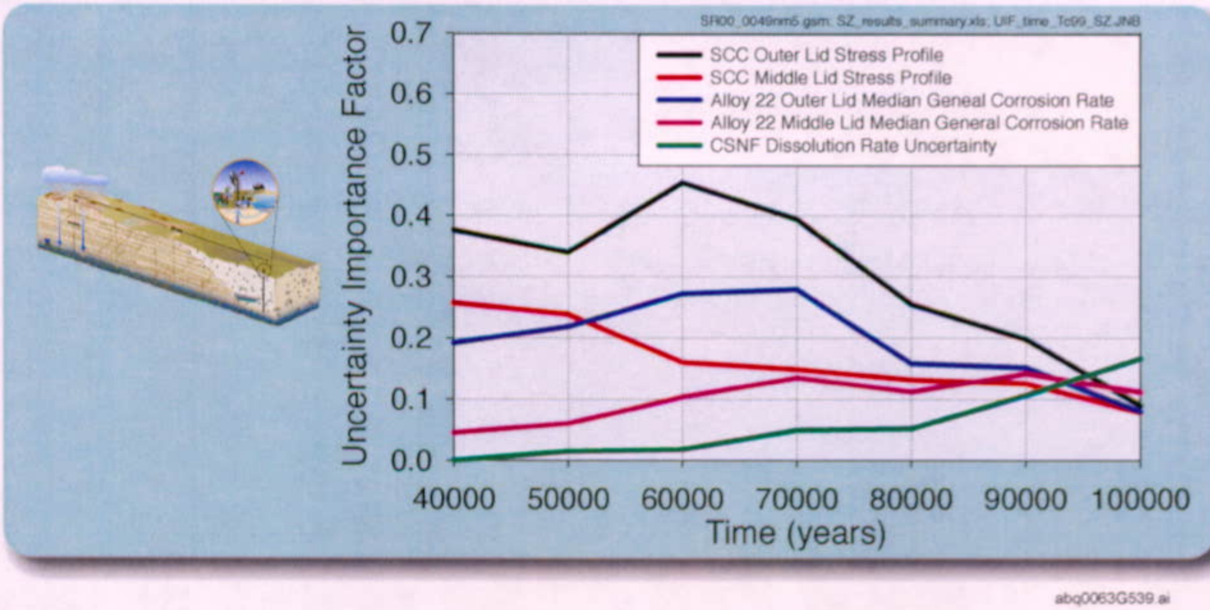

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Figure 5.1-13. Uncertainty Importance Factors at Multiple Time Slices for Mass Release of ^{99}Tc from Saturated Zone, Nominal Scenario

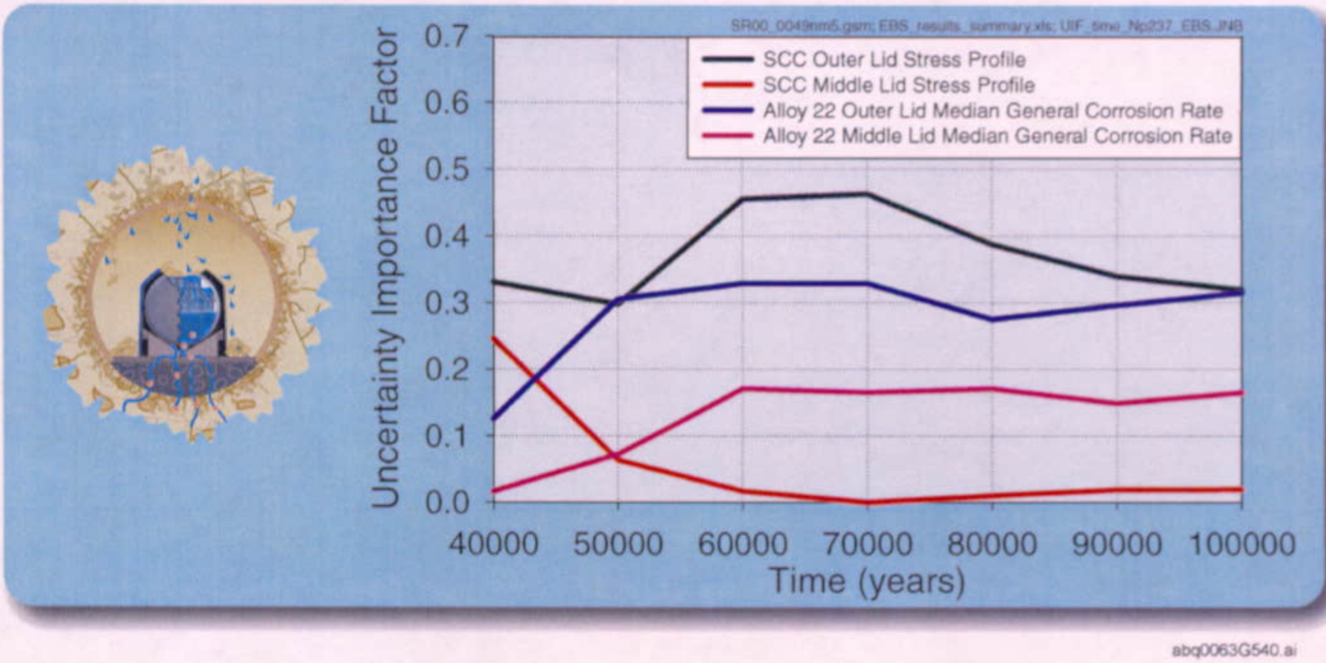

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Figure 5.1-14. Uncertainty Importance Factors at Multiple Time Slices for Mass Release of ^{237}Np from Engineered Barrier System, Nominal Scenario

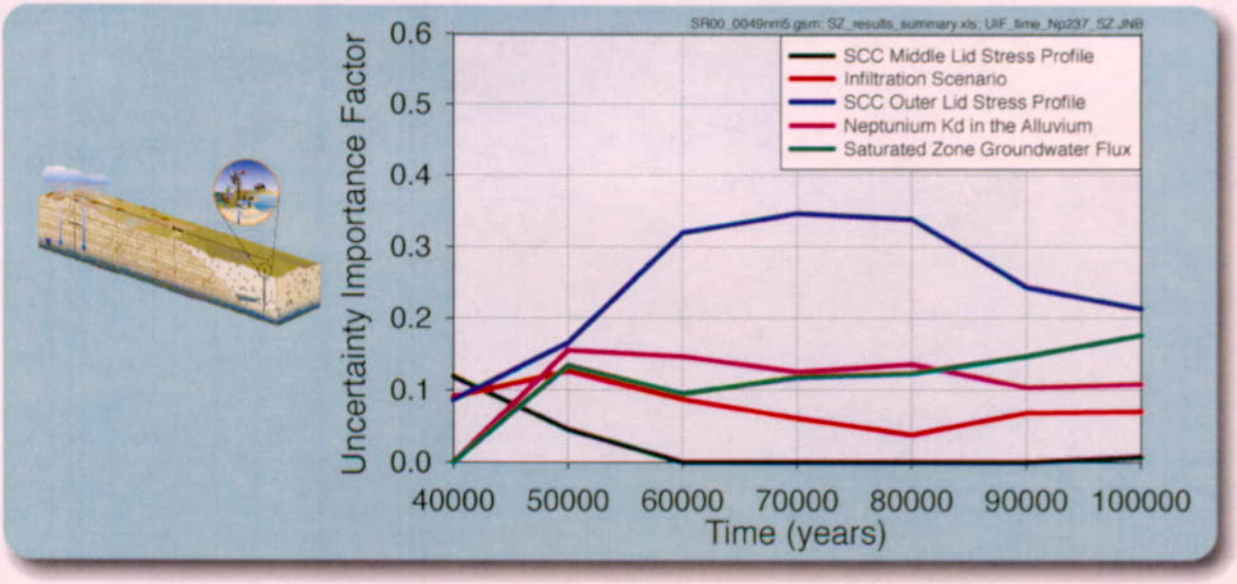

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Figure 5.1-15. Uncertainty Importance Factors at Multiple Time Slices for Mass Release of ^{237}Np from Saturated Zone, Nominal Scenario

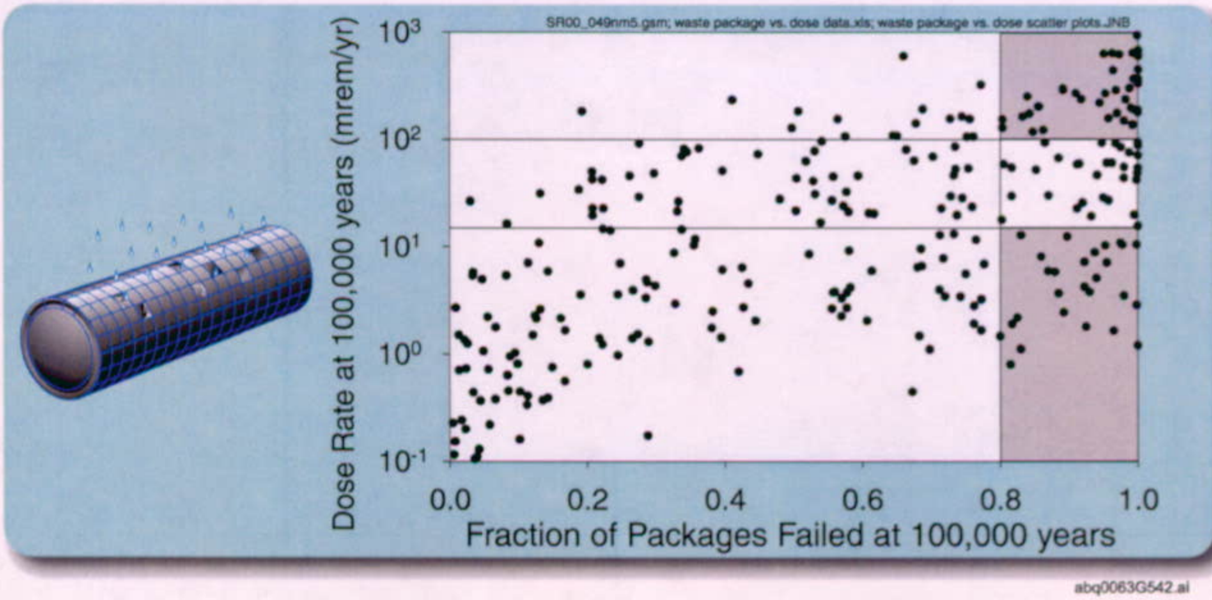

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Figure 5.1-16. Scatter Plot of Total Dose and Fraction Waste Packages Failed at 100,000 Years, Nominal Scenario

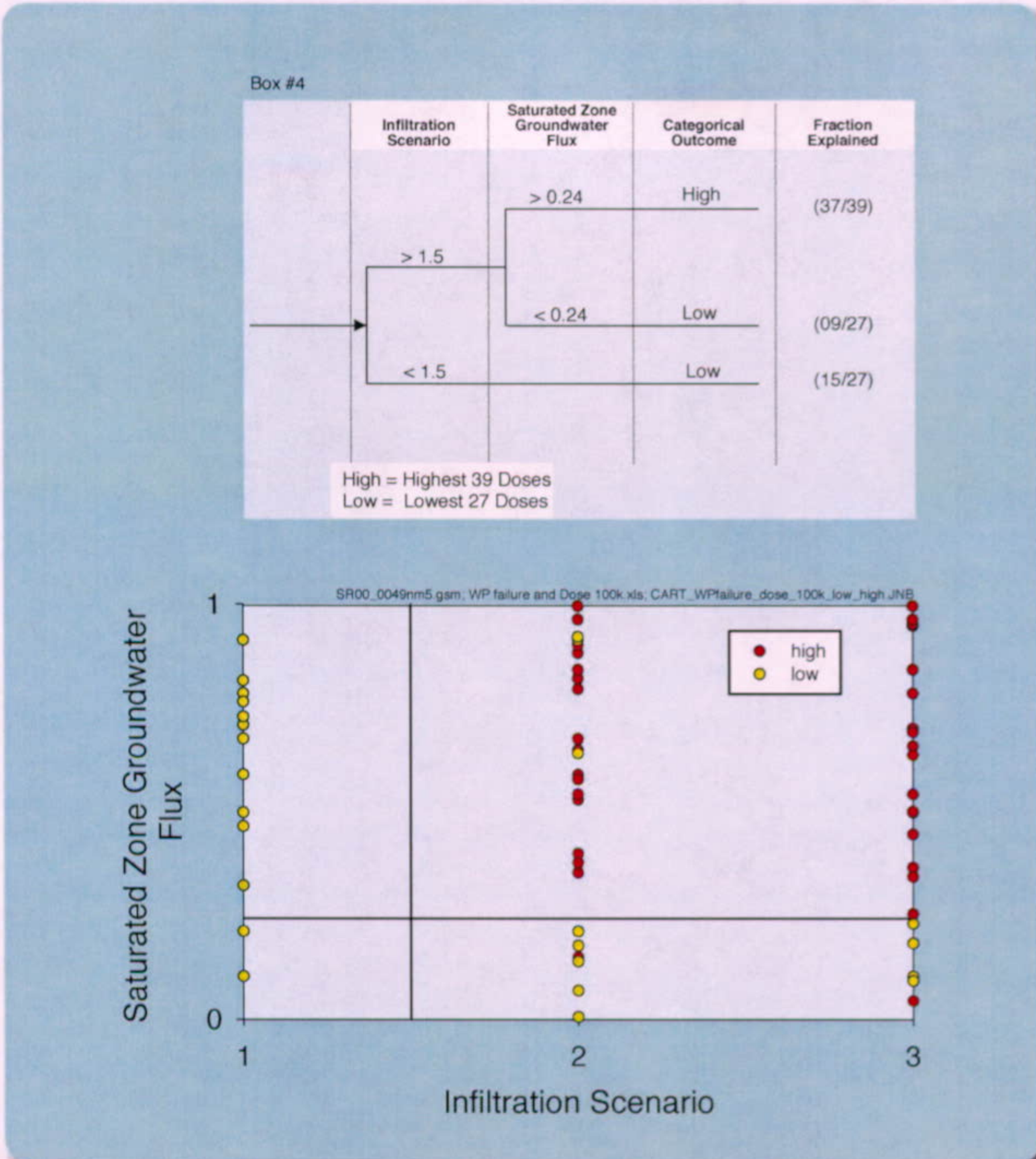

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Figure 5.1-17. Decision Tree Summarizing Classification Tree Analysis (Top) and Partition Plot Showing Clusters of Low-Dose (15 mrem/yr and Lower) and High-Dose (100 mrem/yr and Higher) Outcomes for Realizations with at Least 80 Percent Failed Packages (Bottom) for the Two Most Important Variables at 100,000 Years, Nominal Scenario

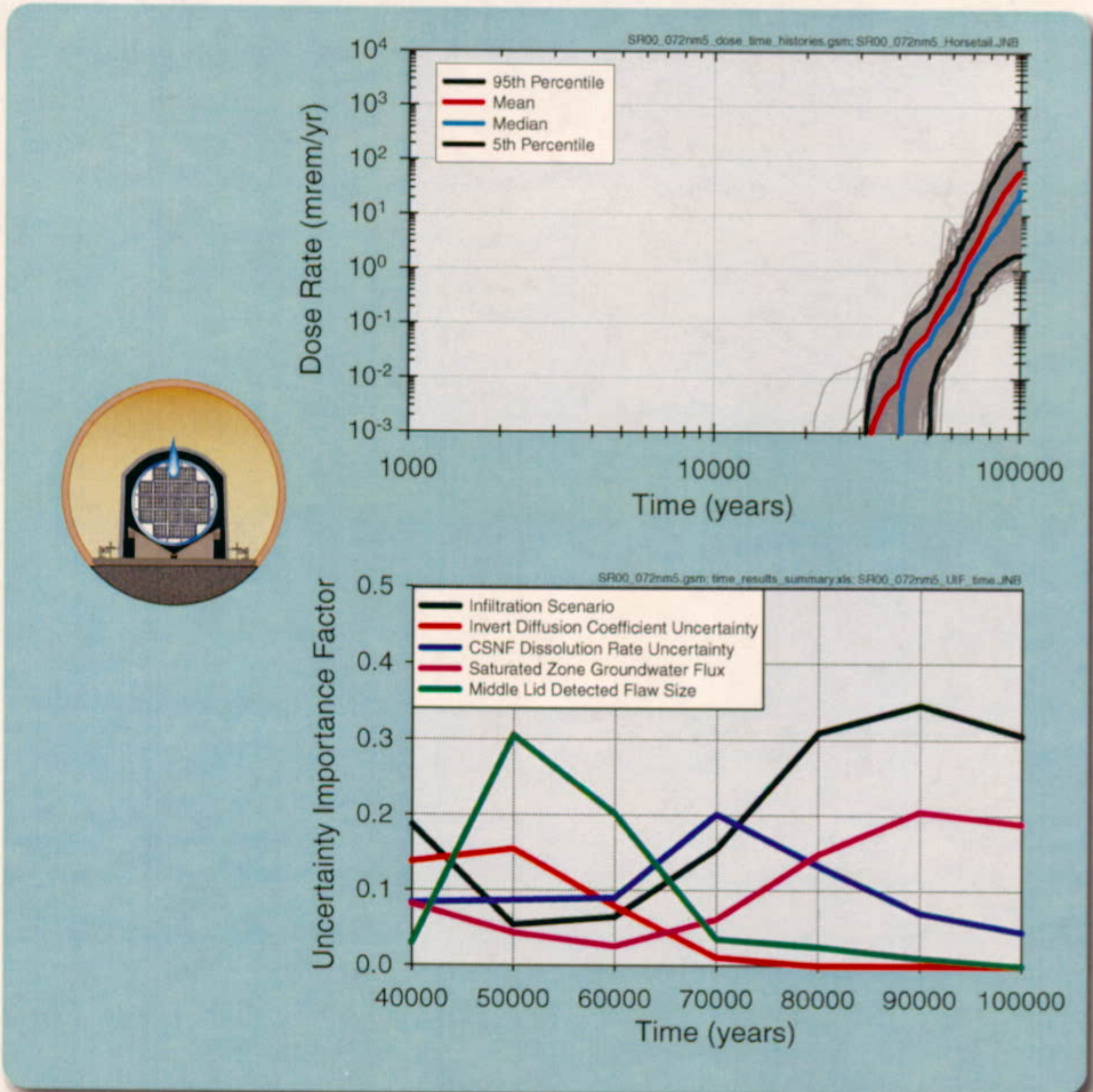

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Figure 5.1-18. Probabilistic Results (Top) and Uncertainty Importance Factors at Multiple Time Slices (Bottom), Nominal Scenario with Key Waste Package Parameters Fixed at Median Values

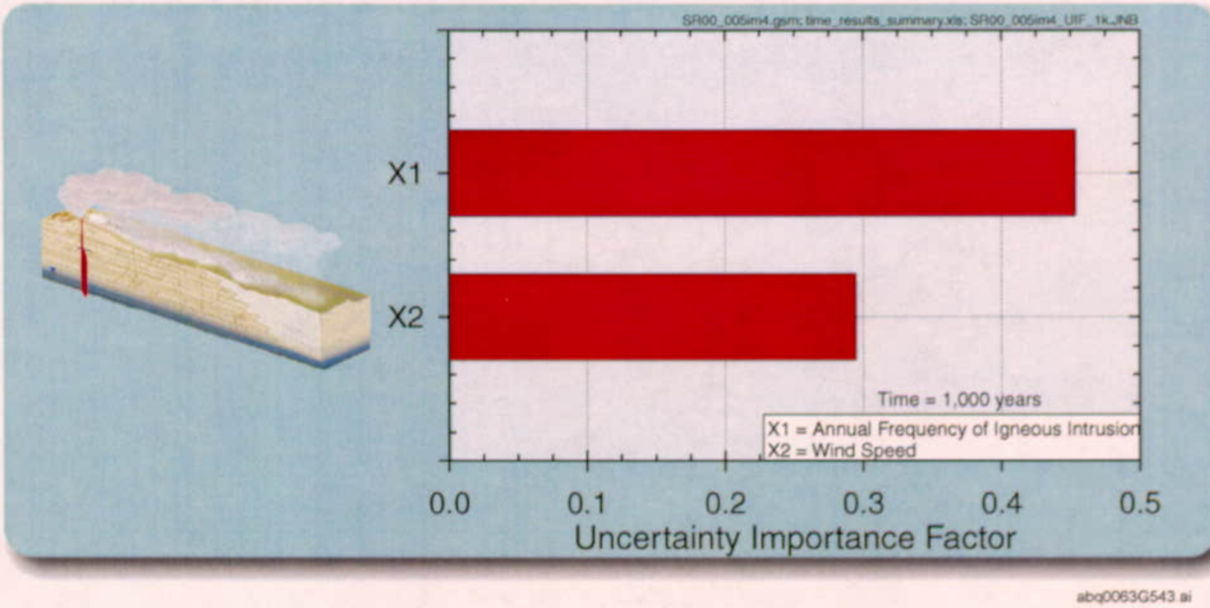

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Figure 5.1-19. Bar Chart Showing Uncertainty Importance Factors for the Most Important Variables at 1,000 Years for Total Dose, Igneous Scenario

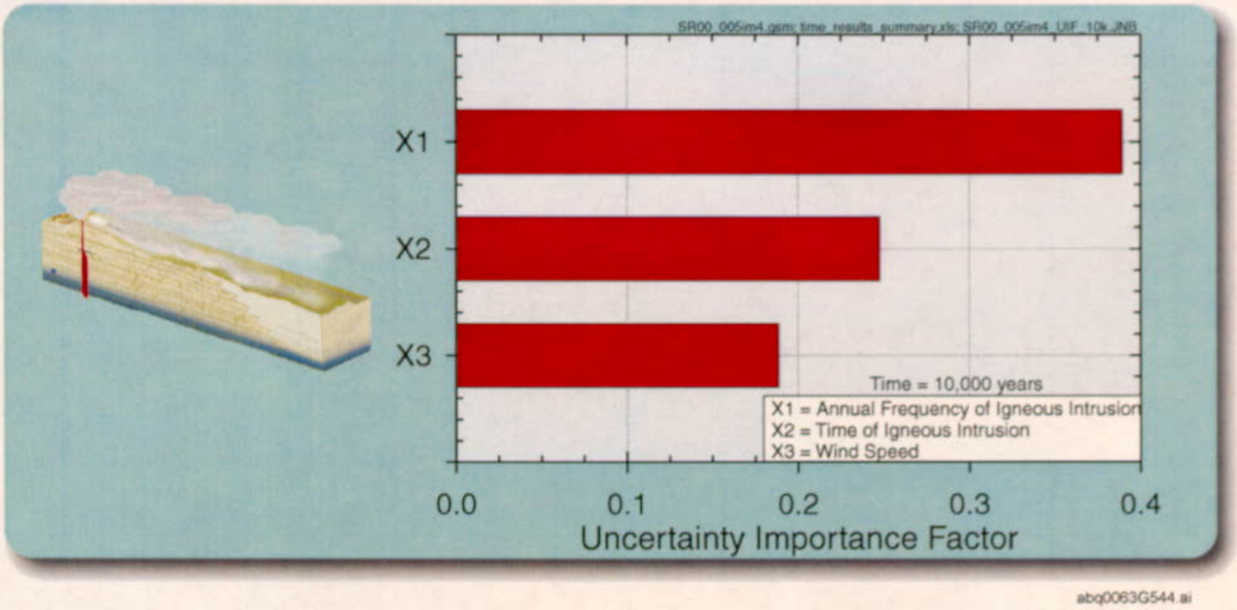

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Figure 5.1-20. Bar Chart Showing Uncertainty Importance Factors for the Most Important Variables at 10,000 Years for Total Dose, Igneous Scenario

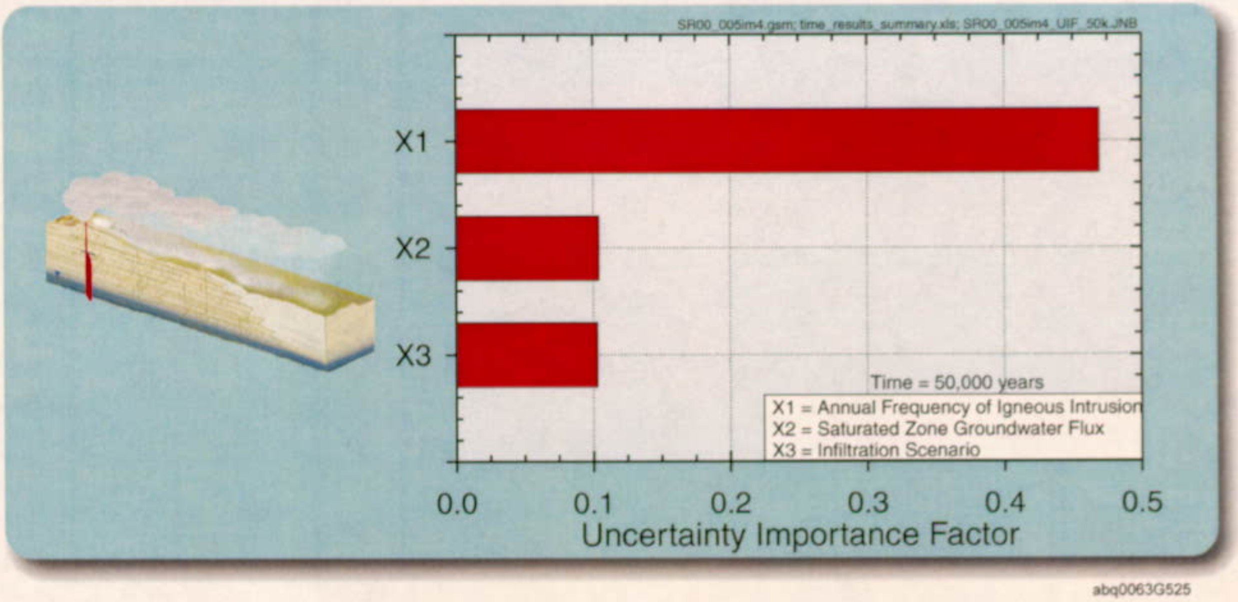

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Figure 5.1-21. Bar Chart Showing Uncertainty Importance Factors for the Most Important Variables at 50,000 Years for Total Dose, Igneous Scenario

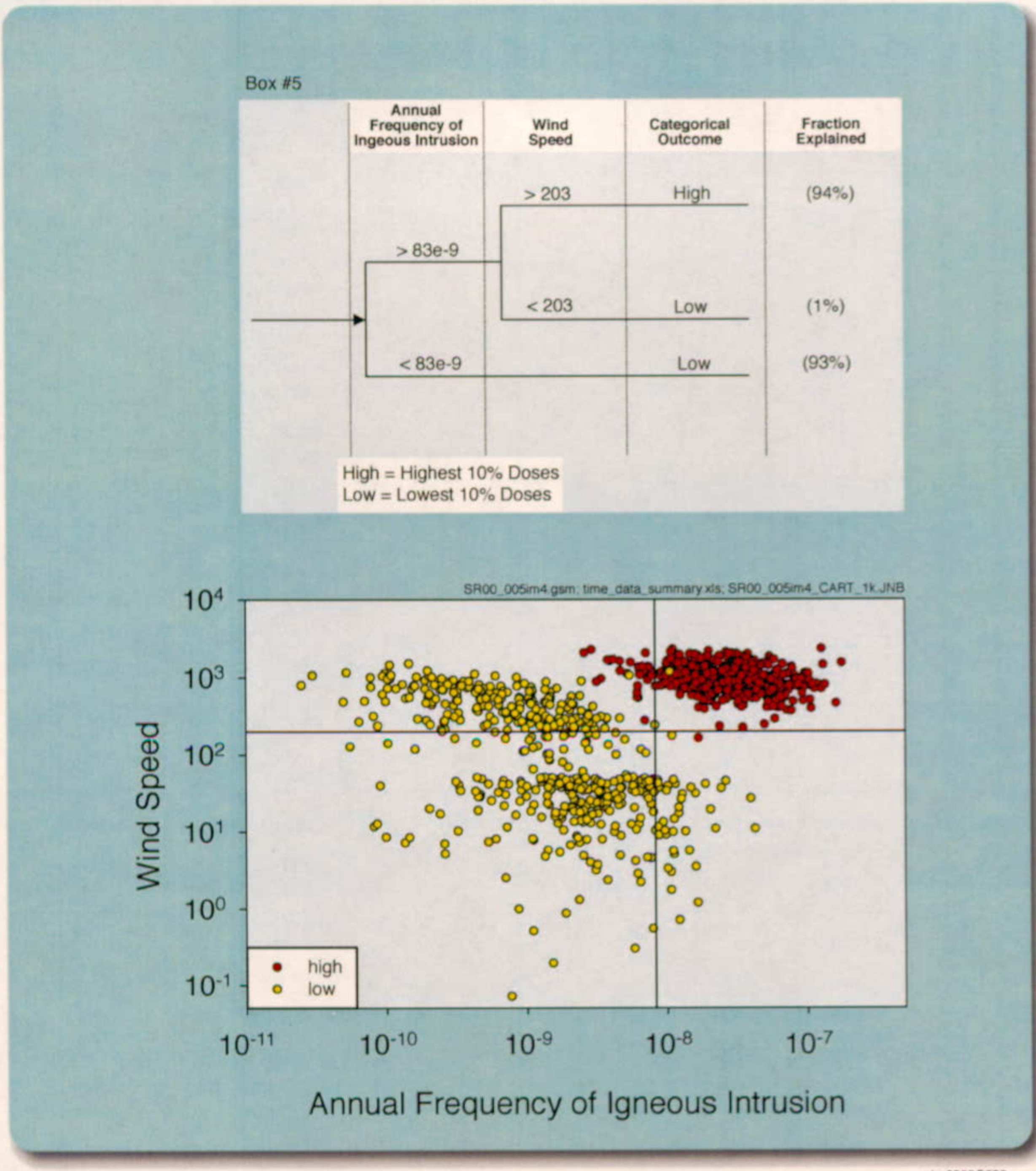

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Figure 5.1-22. Decision Tree Summarizing Classification Tree Analysis (Top) and Partition Plot Showing Clusters of Low-Dose (10th-Percentile and Lower) and High-Dose (90th-Percentile and Higher) Outcomes (Bottom) for the Two Most Important Variables at 1,000 Years, Igneous Scenario

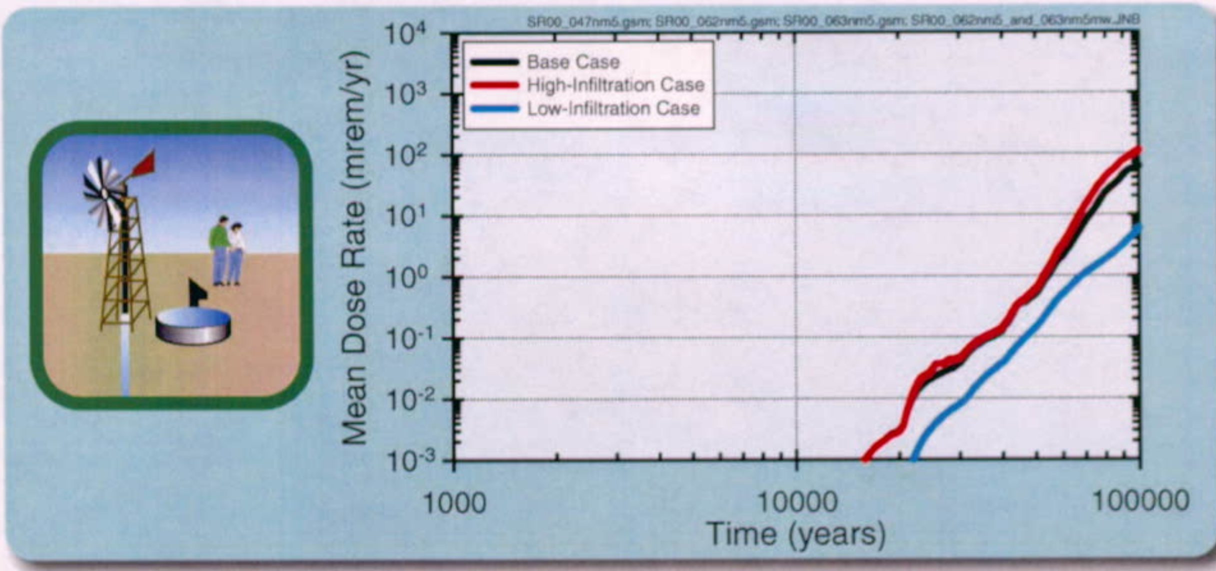

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Figure 5.2-1. Comparison of Mean Dose Rate for High- and Low-Infiltration Cases and Base Case

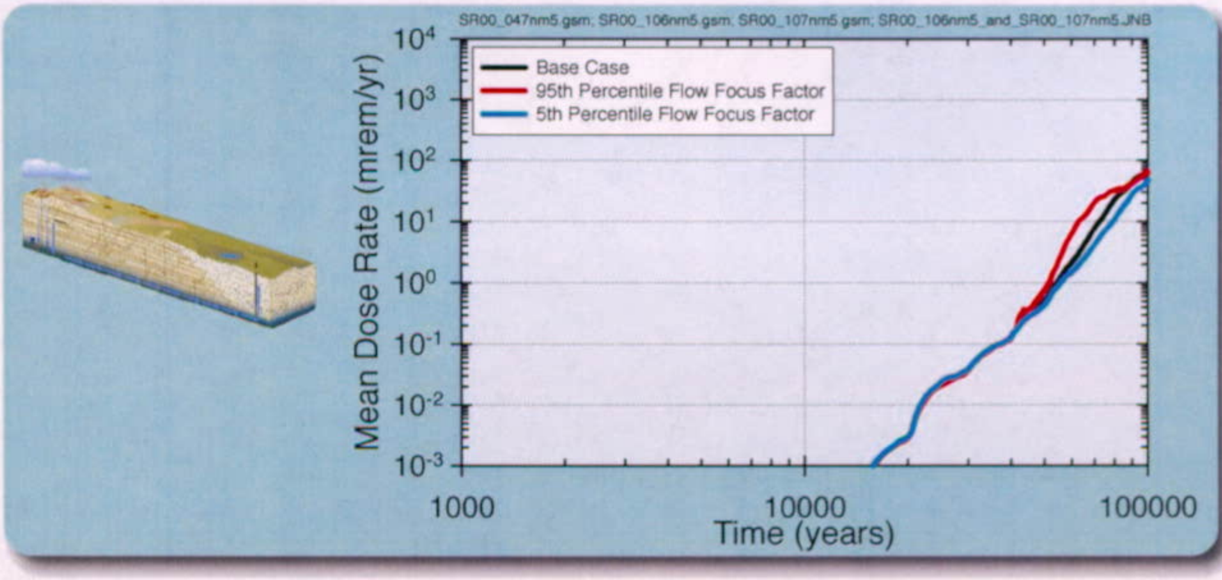

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Figure 5.2-2a. Comparison of Mean Dose Rate for Flow-Focusing Sensitivity Cases and Base Case

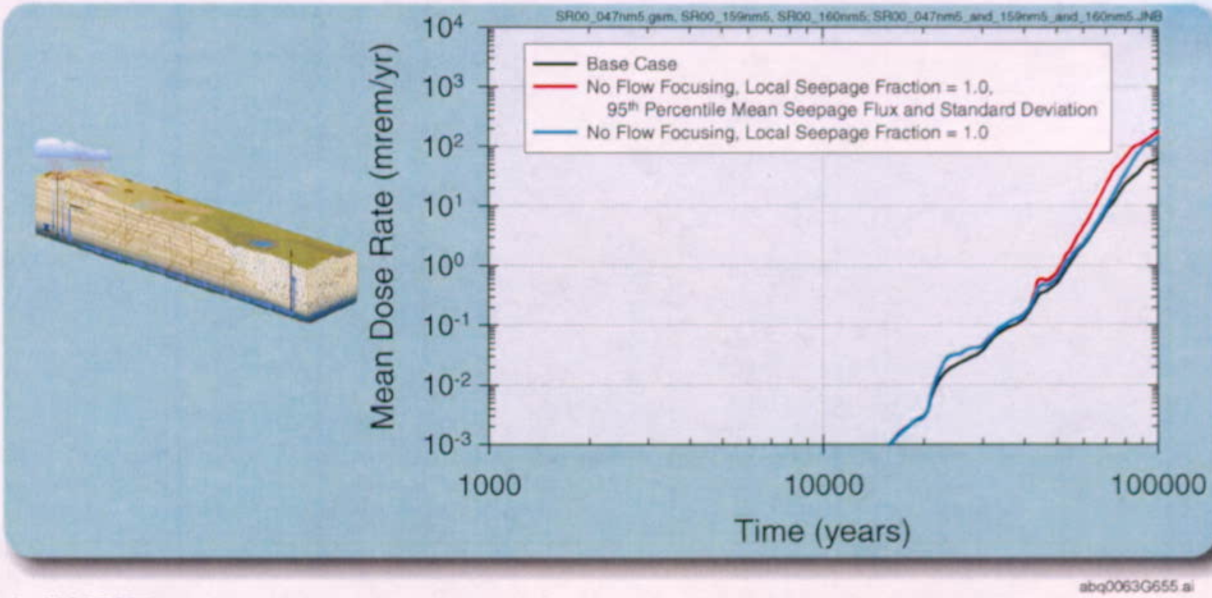

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Figure 5.2-2b. Comparison of Mean Dose Rate for Base Case with Sensitivity Cases having no Flow Focusing and no Local Spatial Variability of the Seepage Fraction

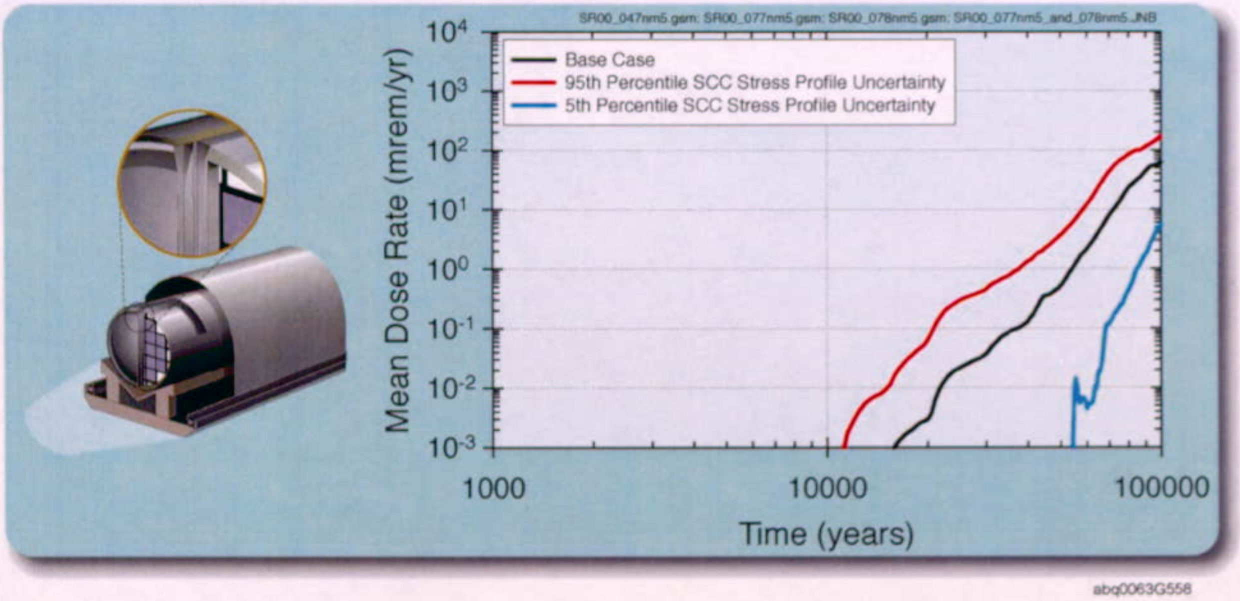

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Figure 5.2-3. Sensitivity of the Mean Dose Rate Profile to the Uncertainty of the Residual Hoop Stress and Stress Intensity Factor in the Outer Lid and Inner Lid Closure Welds

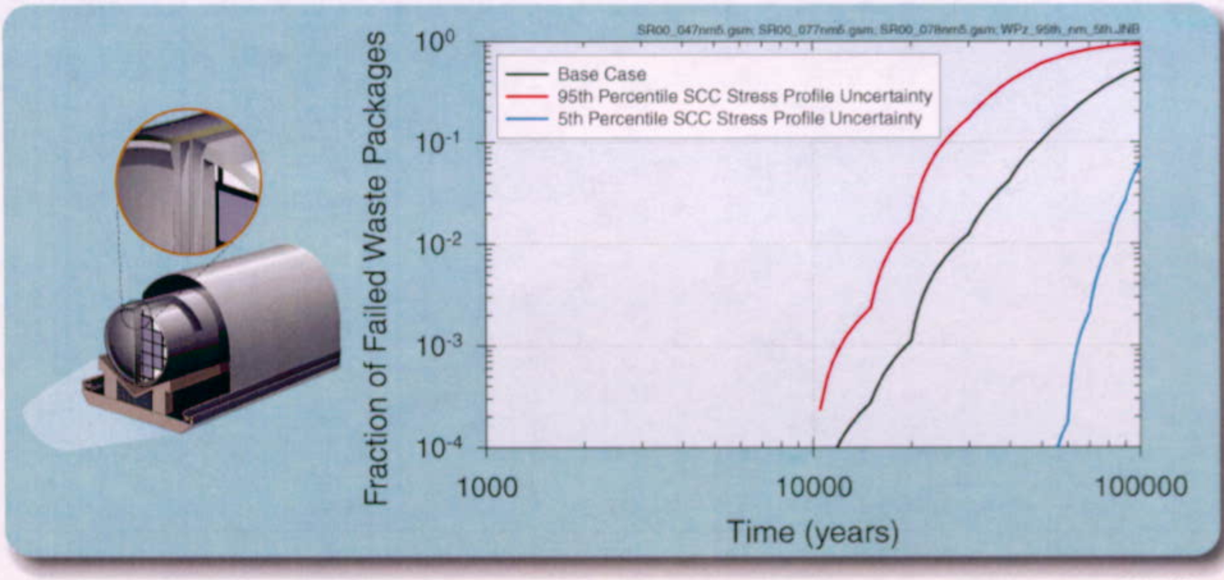

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Figure 5.2-4. Sensitivity of the Mean Waste Package Failure Profile to the Uncertainty of the Residual Hoop Stress and Stress Intensity Factor in the Outer Lid and Inner Lid Closure Welds

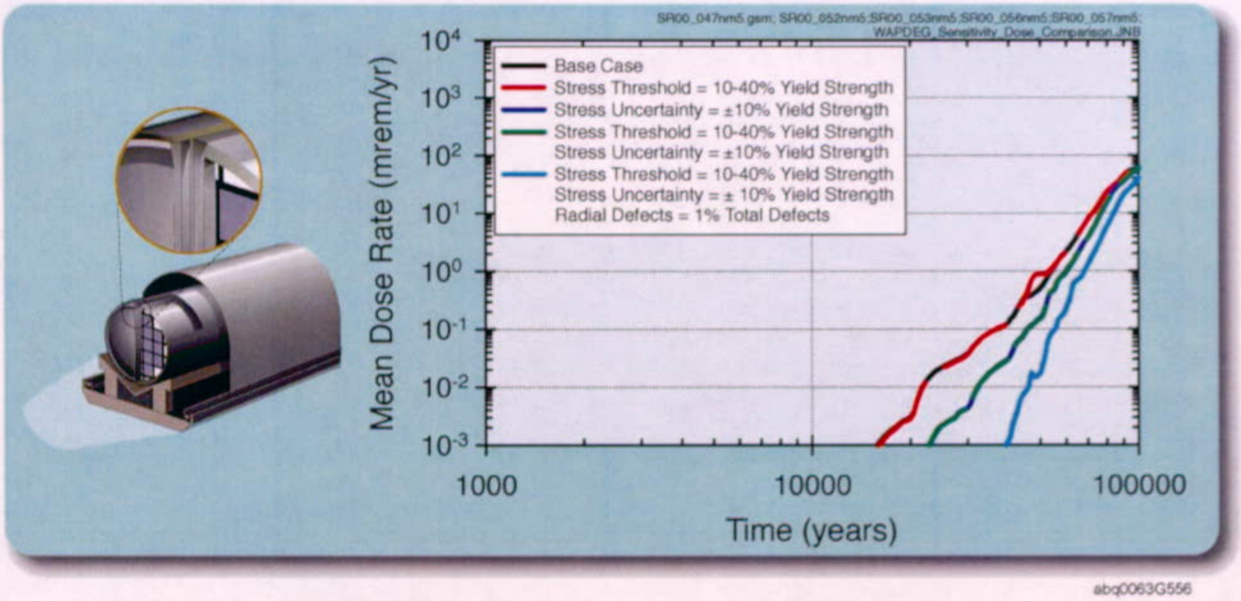

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Figure 5.2-5. Sensitivity of the Mean Dose Rate Profile to Alternative Uncertainty Ranges of the Stress Corrosion Cracking Model Parameters for the Waste Package Closure-Lid Welds

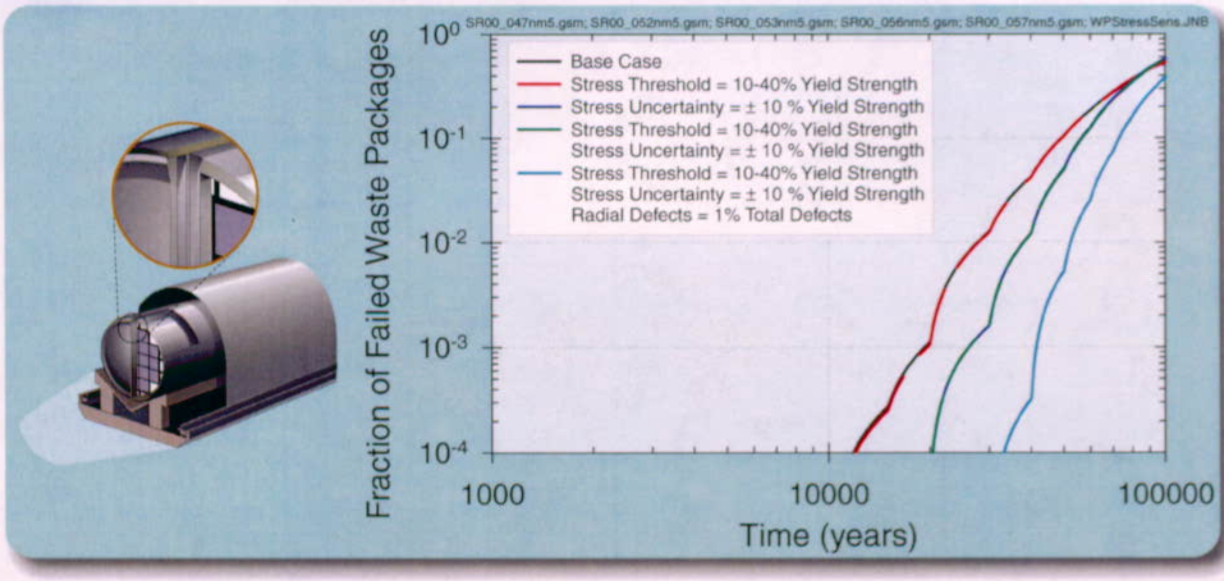

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Figure 5.2-6. Sensitivity of the Mean Waste Package Failure Profile to Alternative Uncertainty Ranges of the Stress Corrosion Cracking Model Parameters for the Waste Package Closure-Lid Welds

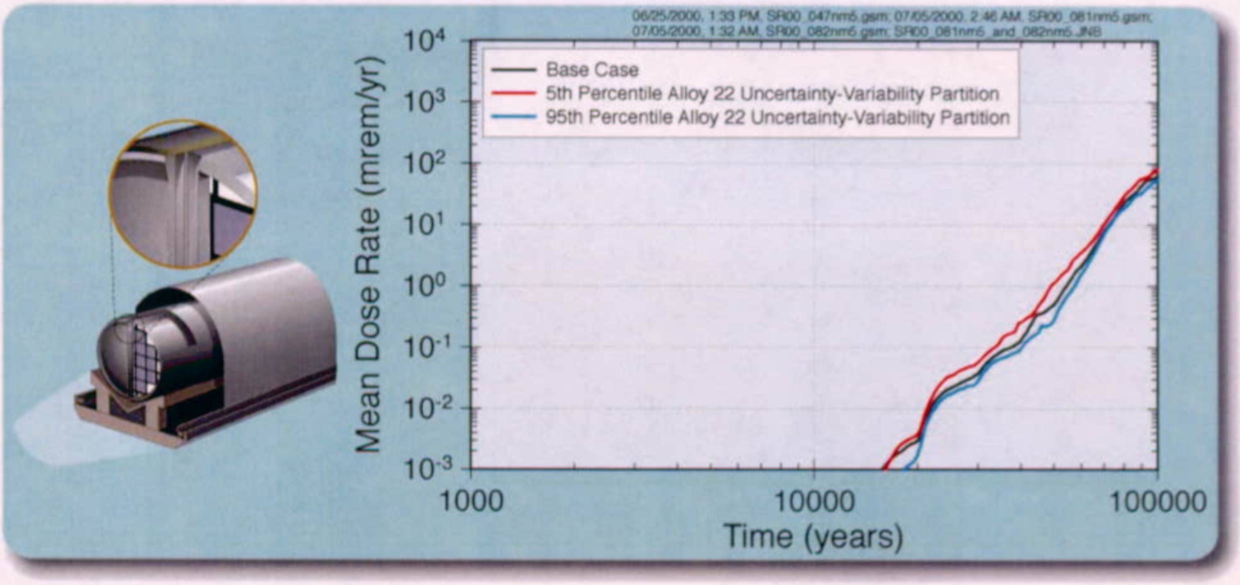

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Figure 5.2-7. Sensitivity of the Mean Dose Rate Profile to the Uncertainty and Variability Partitioning Ratio for Alloy-22 General Corrosion Rate

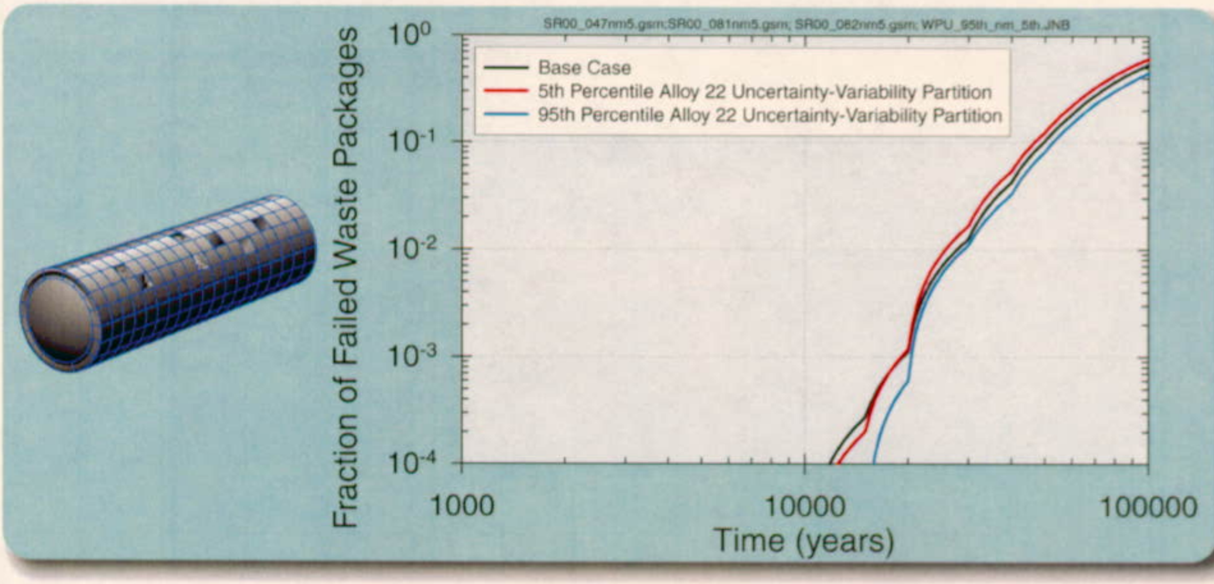

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Figure 5.2-8. Sensitivity of the Mean Waste Package Failure Profile to the Uncertainty and Variability Partitioning Ratio for Alloy-22 General Corrosion Rate

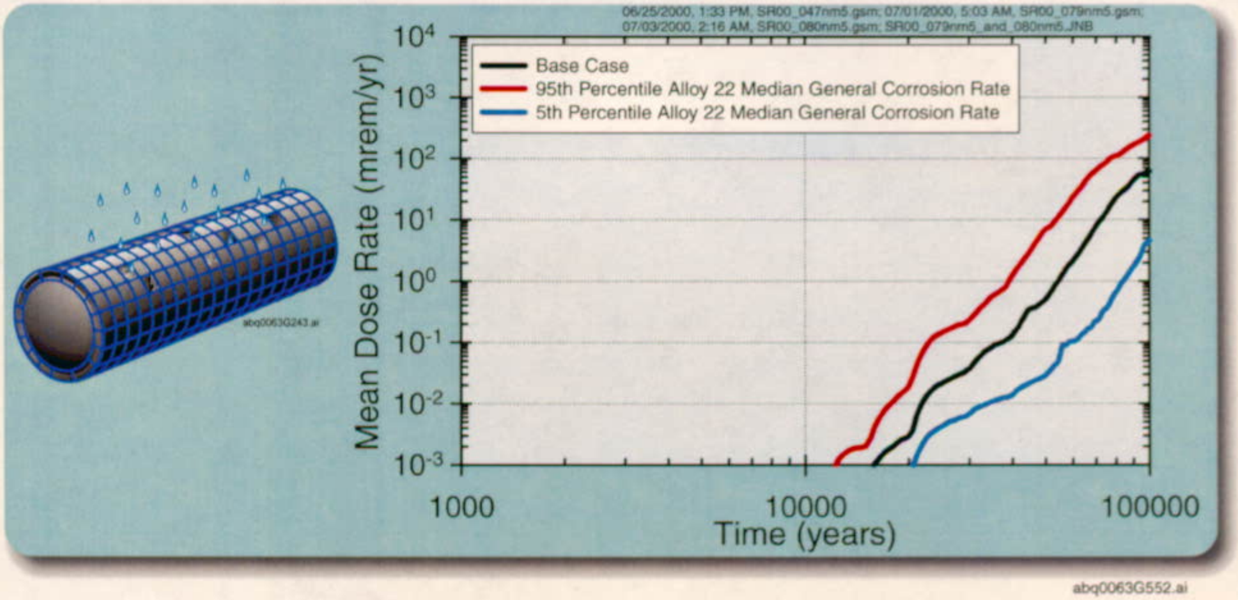

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Figure 5.2-9. Sensitivity of the Mean Dose Rate Profile to the Median General Corrosion Rate of Alloy-22

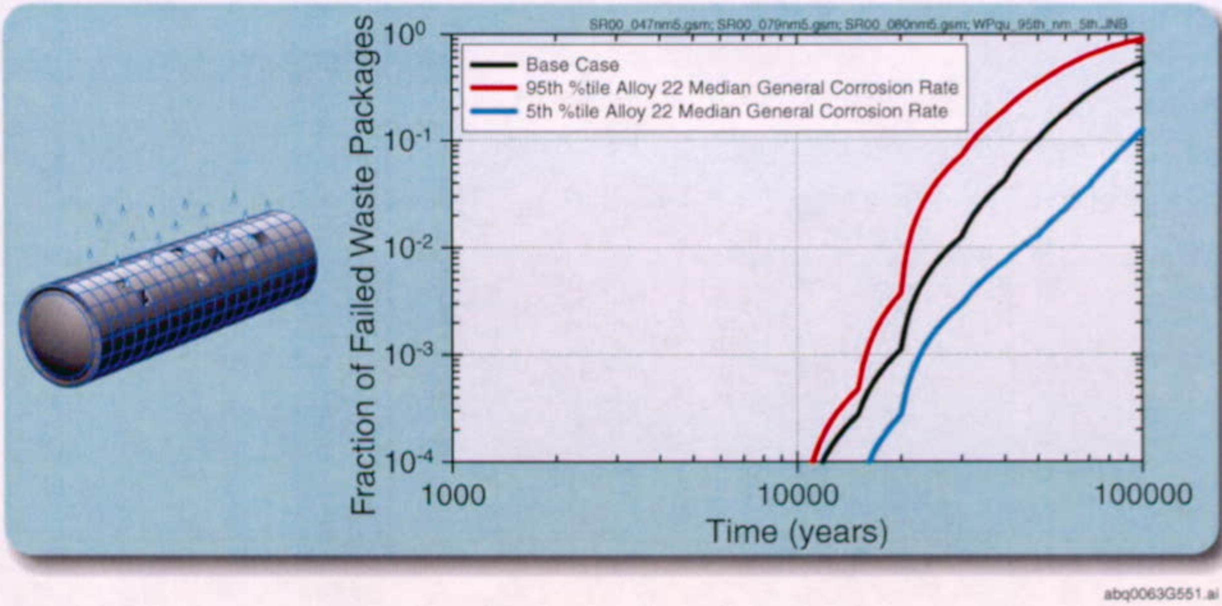

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Figure 5.2-10. Sensitivity of the Mean Waste Package Failure Profile to the Median General Corrosion Rate of Alloy-22

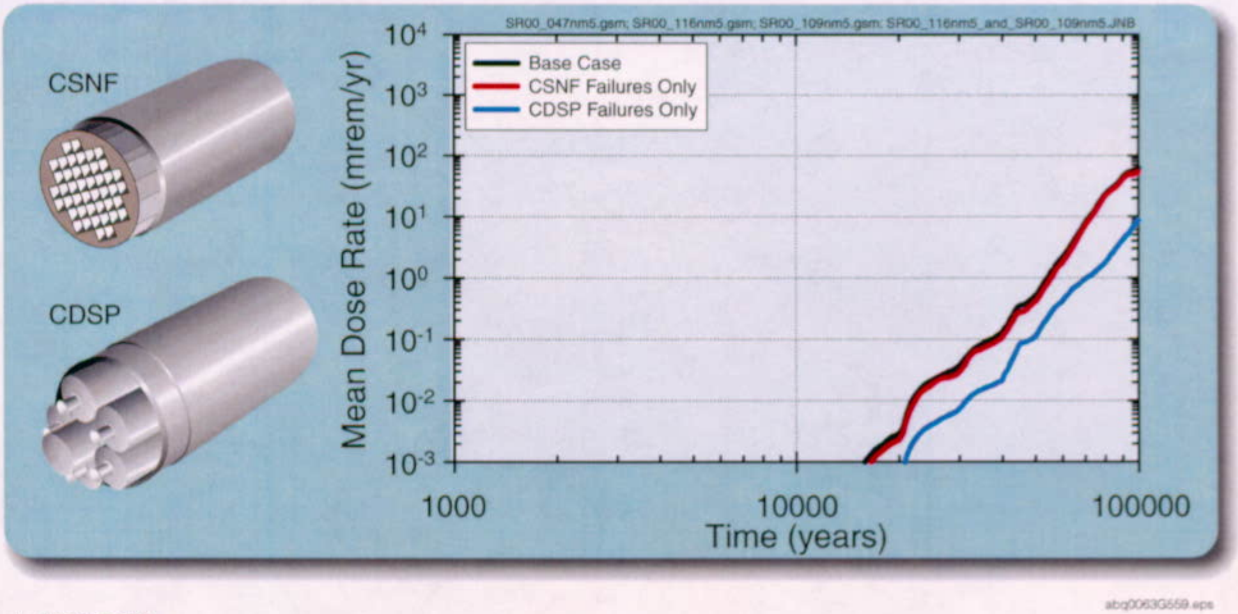

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Figure 5.2-11. Mean Base Case Dose Rate Compared to Cases with Commercial Spent Nuclear Fuel Package Failures only and Co-Disposal Package Failures only

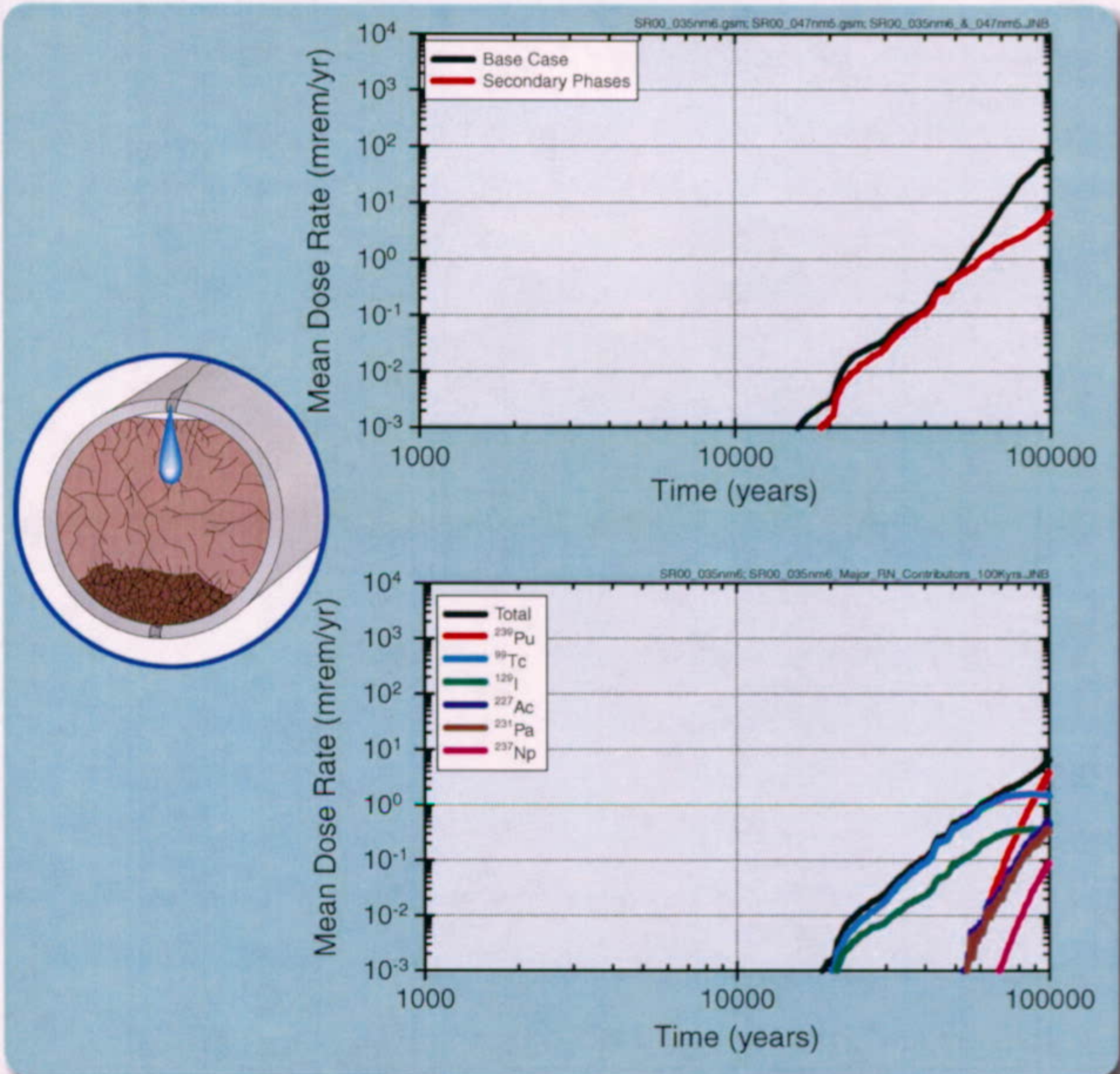

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Figure 5.2-12. Mean Dose Rate of Major Radionuclide Contributors for the Secondary Mineral Phase Sensitivity Case

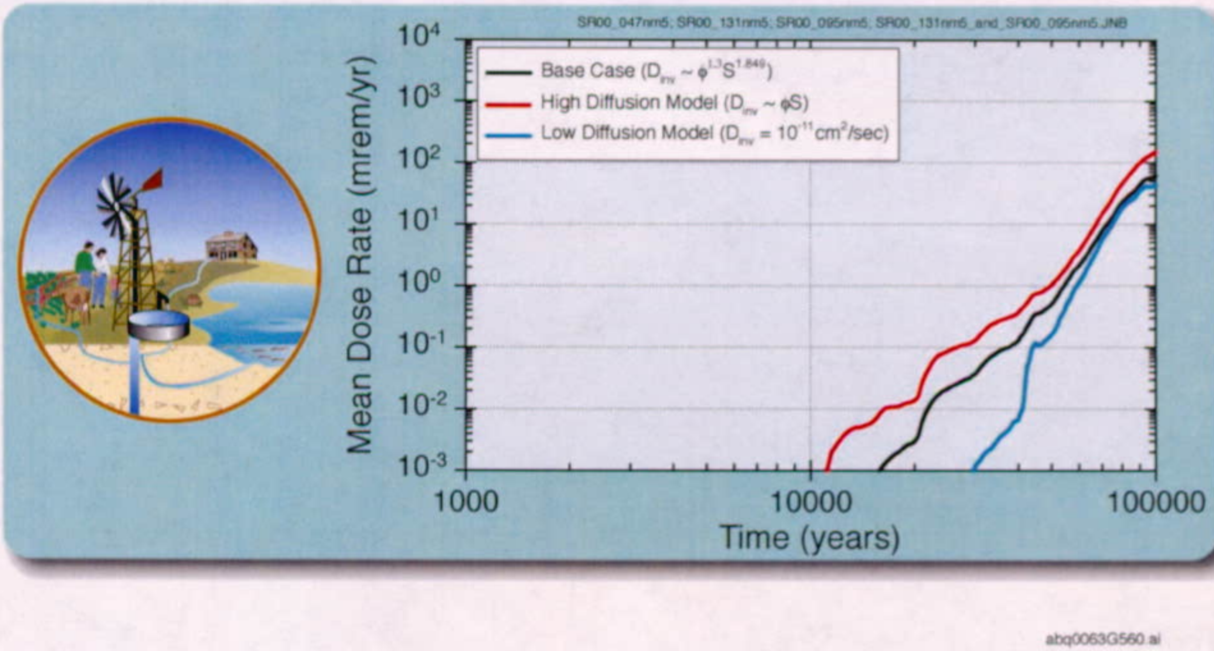

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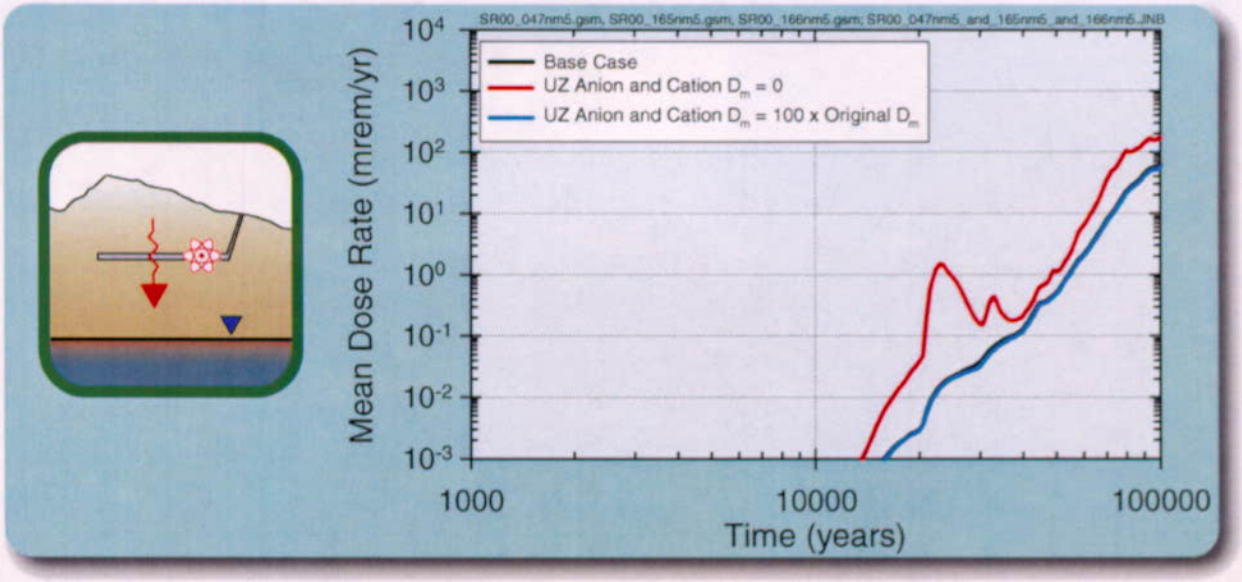
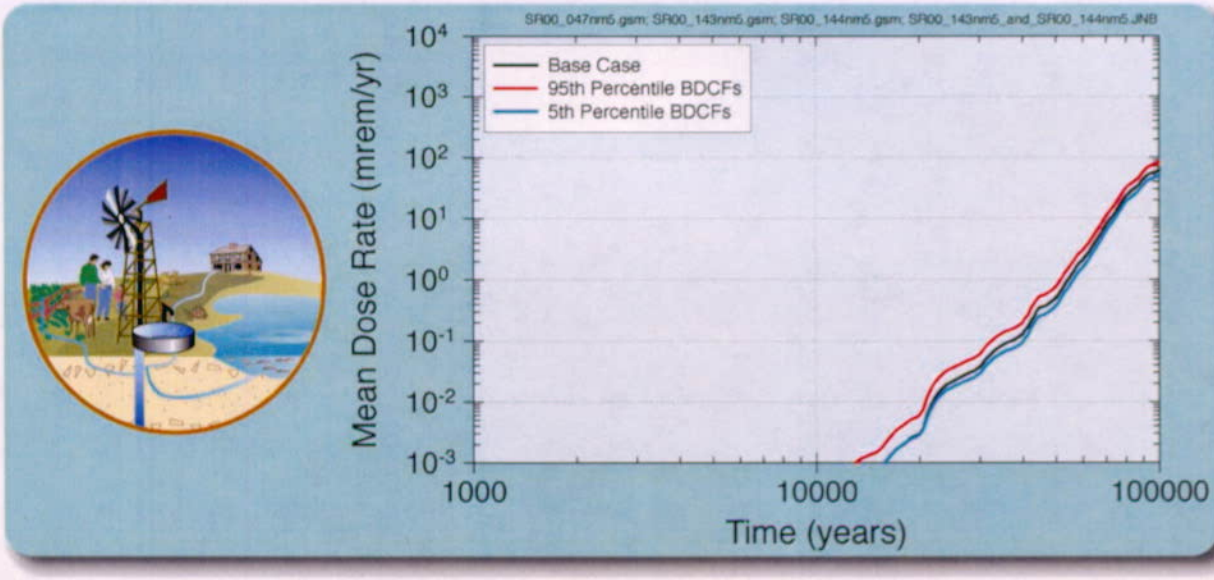
Figure 5.2-13. Comparison of Mean Dose Rates for Three Invert Diffusion Models



abq0063G652

Figure 5.2-14. Sensitivity to Matrix Diffusion in the Unsaturated Zone



abq0063G562

Figure 5.2-15. Sensitivity to Biosphere Dose Conversion Factors

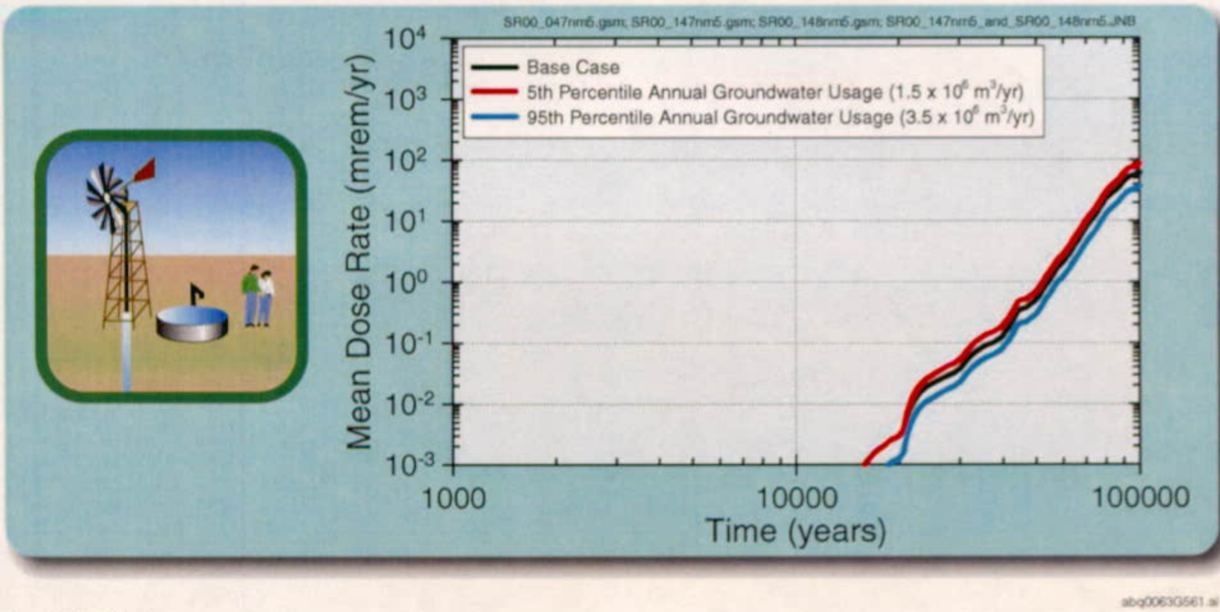


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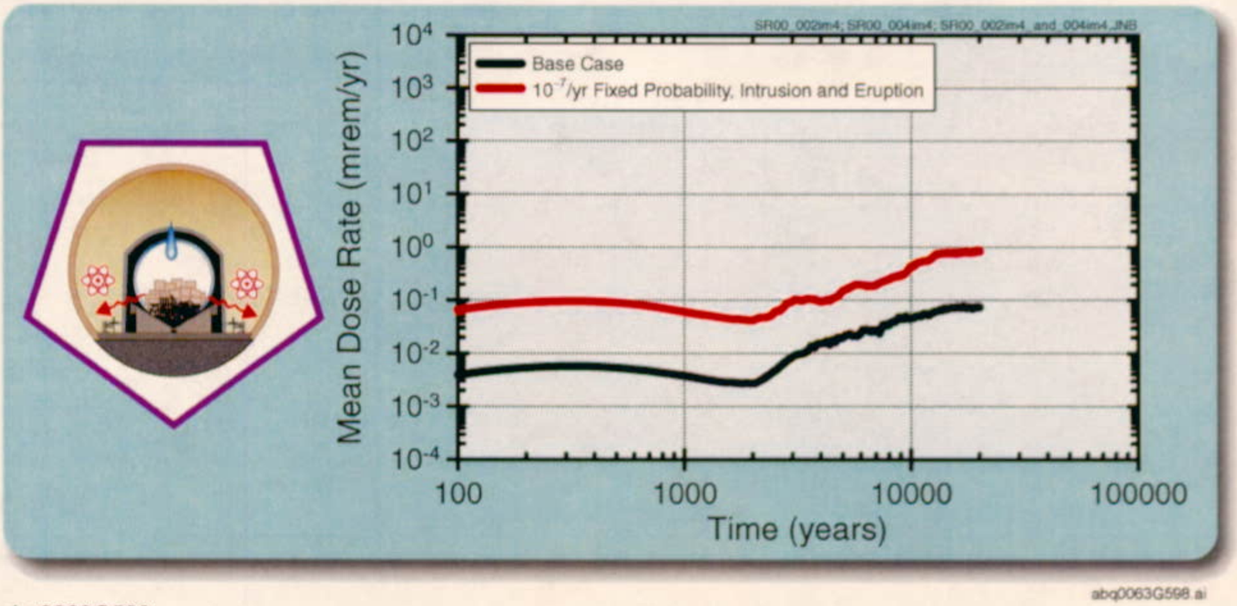
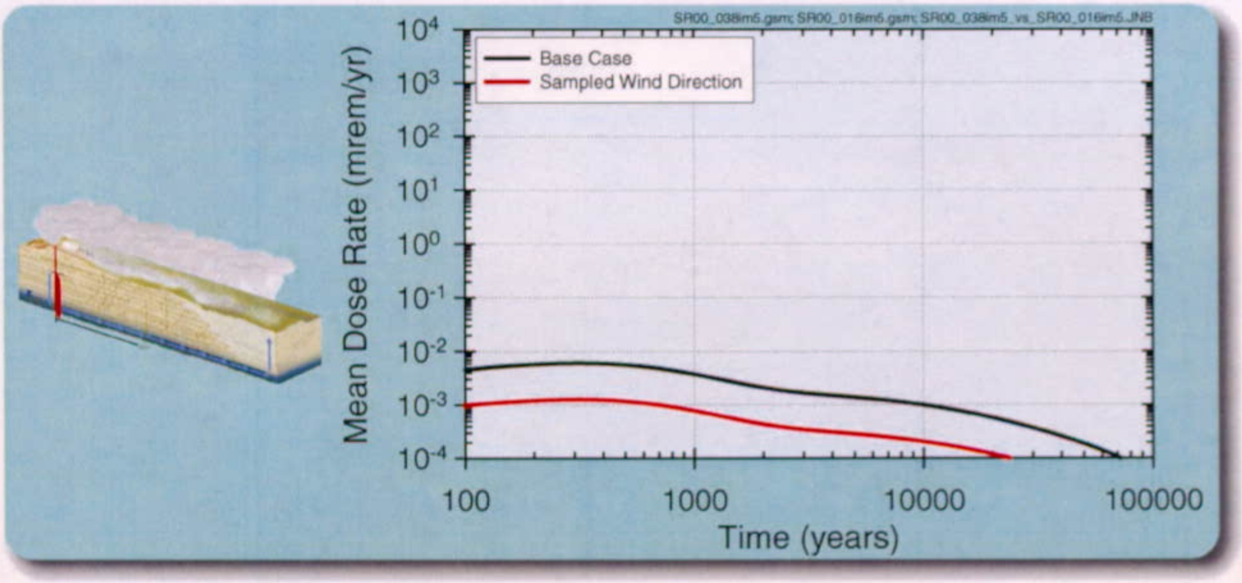
Figure 5.2-16. Sensitivity to Groundwater Usage Volume



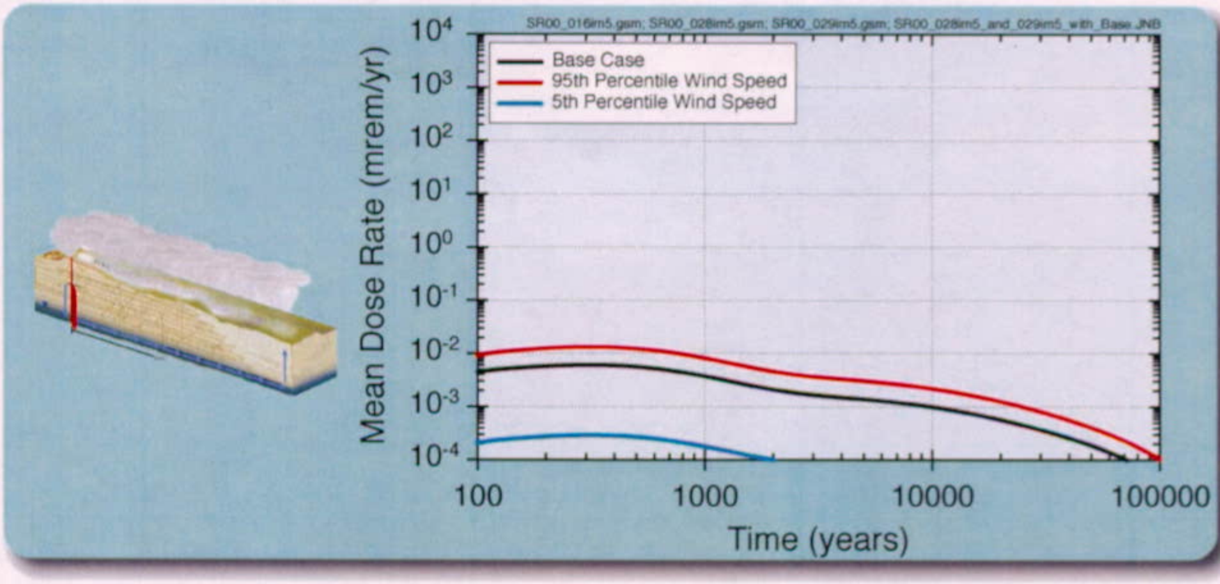
abq0063G598

Figure 5.2-17. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Igneous Dose Rate with the Dose Rate Calculated using a Fixed Annual Probability of Igneous Intrusion and Eruption Equal to 10^{-7}



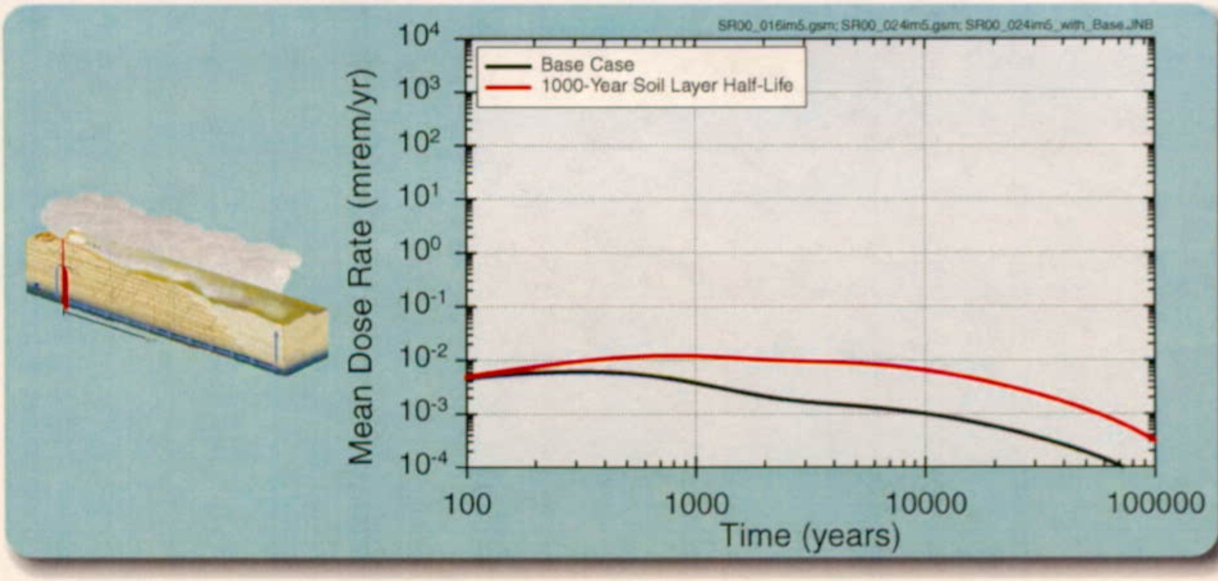
abq0063G603

Figure 5.2-18. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Eruptive Dose Rate with the Dose Rate Calculated using a Sampled Wind Direction, Rather than Assuming that the Wind always Blows Toward the Location of the Critical Group



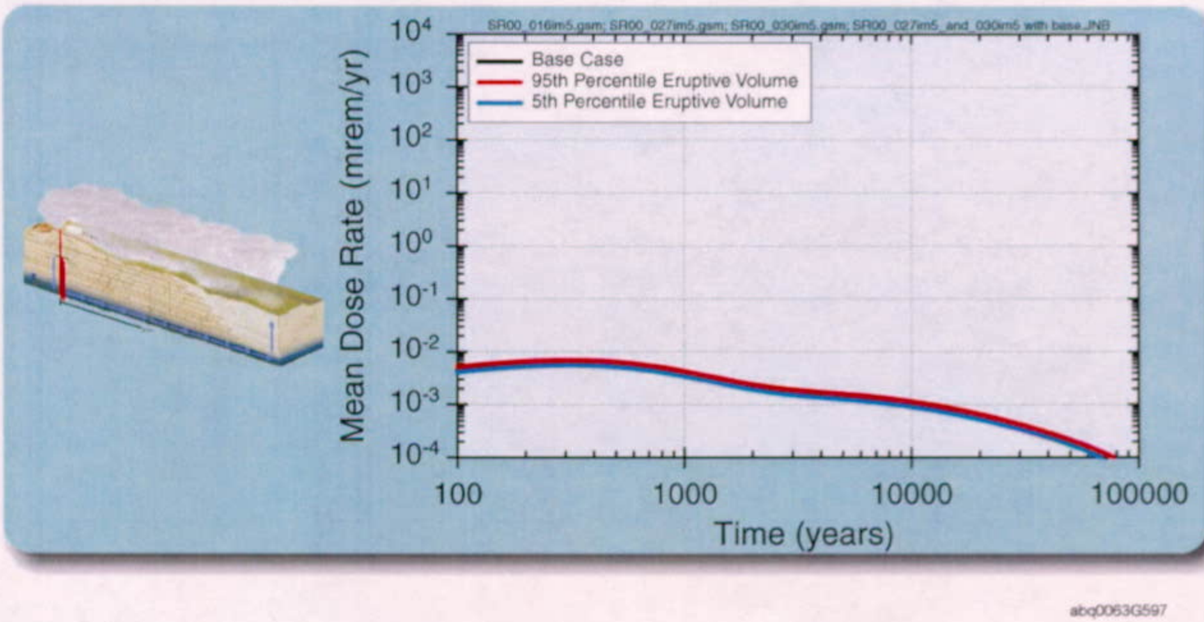
abq0063G602

Figure 5.2-19. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Eruptive Dose Rate with the Dose Rate Calculated using Wind Speed Fixed at the 5th and 95th-Percentile Values from the Distribution



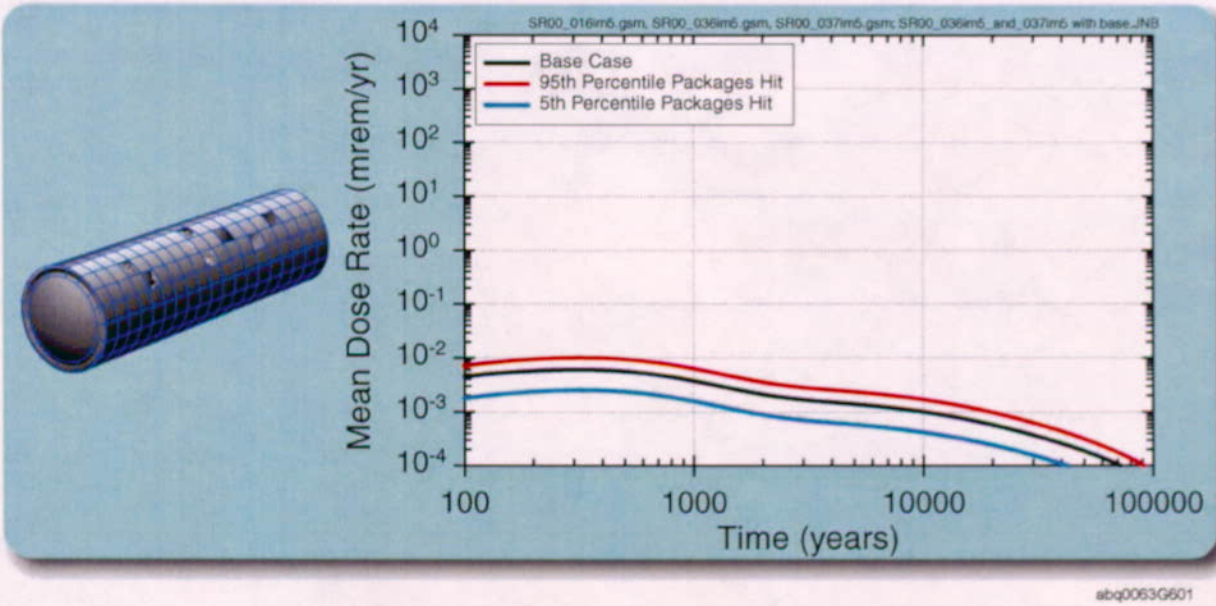
abq0063G605

Figure 5.2-20. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Eruptive Dose Rate with the Dose Rate Calculated using a 1,000-Year Mean Soil Removal Time Following Deposition of the Ash Layer



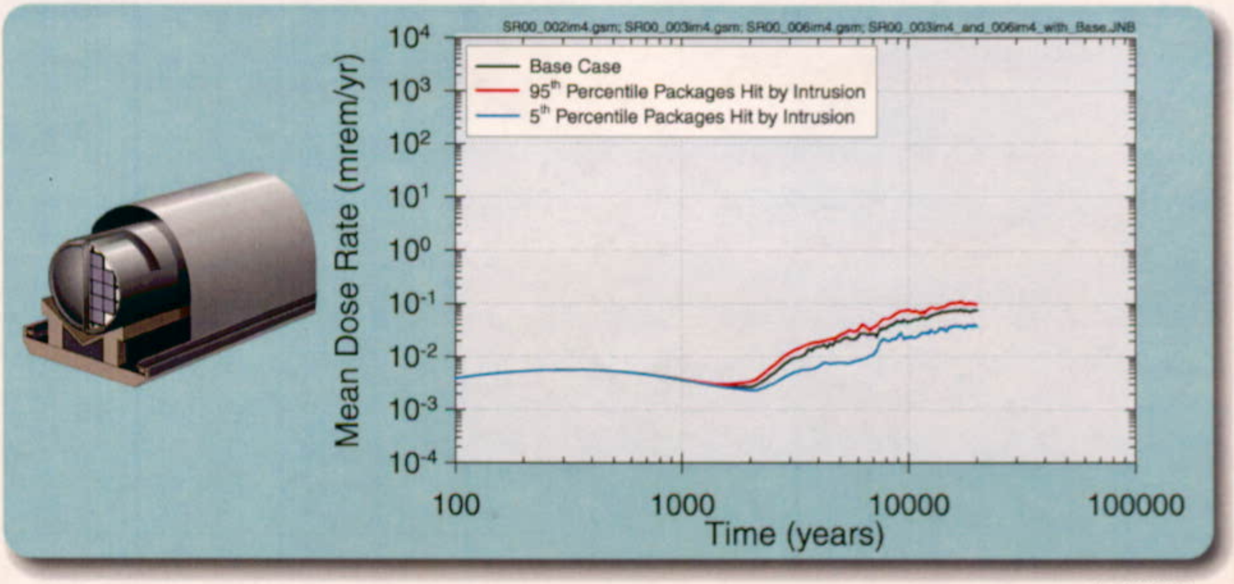
abq0063G597

Figure 5.2-21. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Igneous Dose Rate with the Dose Rate Calculated using the Volume of Erupted Material (which Characterizes the Power of the Eruptive Event) is Fixed at the 5th and 95th-Percentiles of the Distribution used in the TSPA-SR



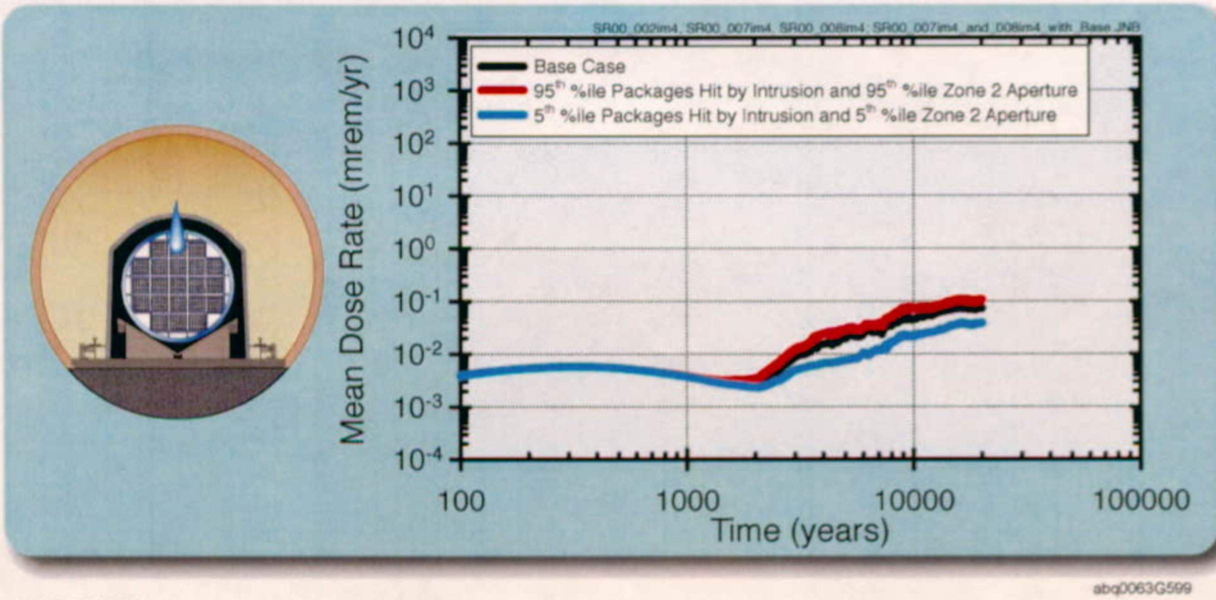
abq0063G601

Figure 5.2-22. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Eruptive Dose Rate with the Dose Rate Calculated Assuming that the Number of Packages Damaged for the Direct Release Event is Fixed at the 5th and 95th-Percentiles of the Distribution used in the TSPA-SR



abq0063G600

Figure 5.2-23. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Igneous Dose Rate with the Dose Rate Calculated Assuming that the Number of Waste Packages Damaged by Intrusion is Fixed at the 5th and 95th-Percentiles of the Distribution used in the TSPA-SR



abq0063G599

Figure 5.2-24. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Igneous Dose Rate with the Dose Rate Calculated Assuming that the Diameter of the Aperture in Waste Packages Damaged by Intrusion and the Number of Waste Packages Damaged in Zones 1 and 2 by Intrusion are Fixed at the 5th and 95th-Percentiles of the Distributions used in the TSPA-SR

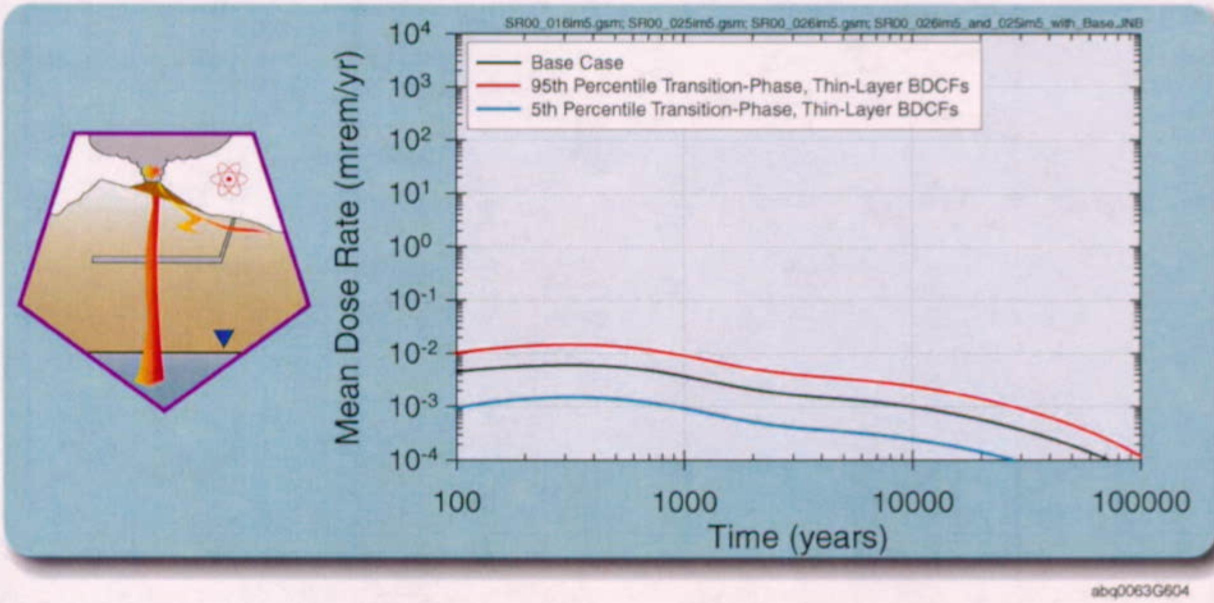


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Figure 5.2-25. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Eruptive Dose Rate with the Dose Rate Calculated Assuming that the Transition-Phase Thin-Layer BDCF's are Fixed at the 5th and 95th-Percentiles of the Distributions used in the TSPA-SR

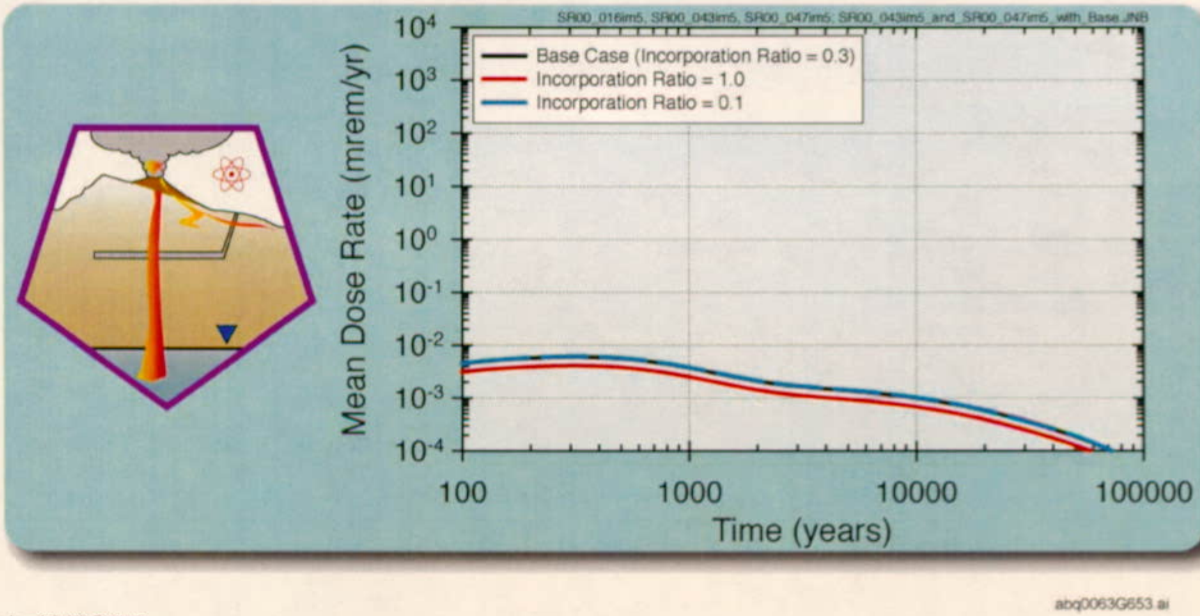

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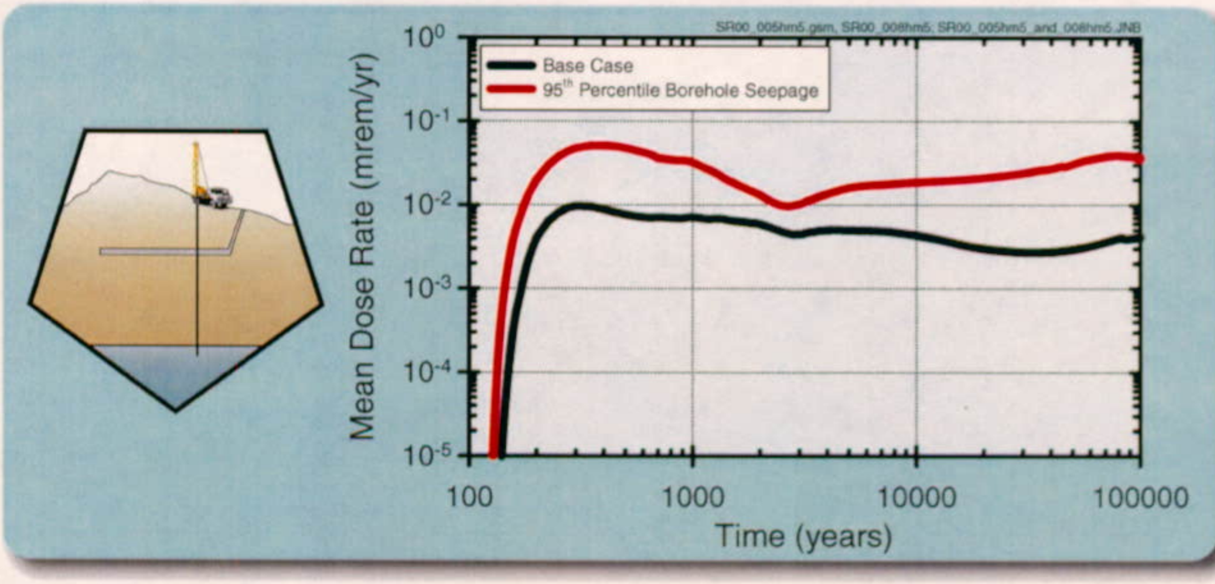
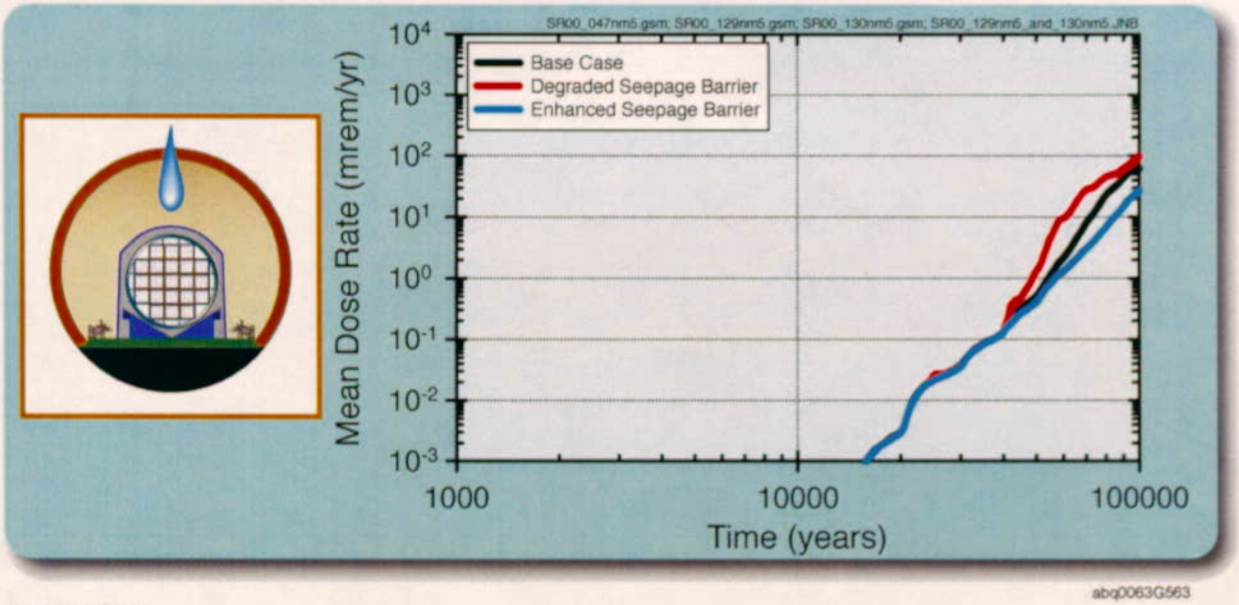
Figure 5.2-26. Comparison of Total System Performance Assessment-Site Recommendation Probability-Weighted Mean Annual Eruptive Dose Rate with Dose Rates Calculated Using Incorporation Ratios of 0.3 (Base Case), 0.1, and 1.0.



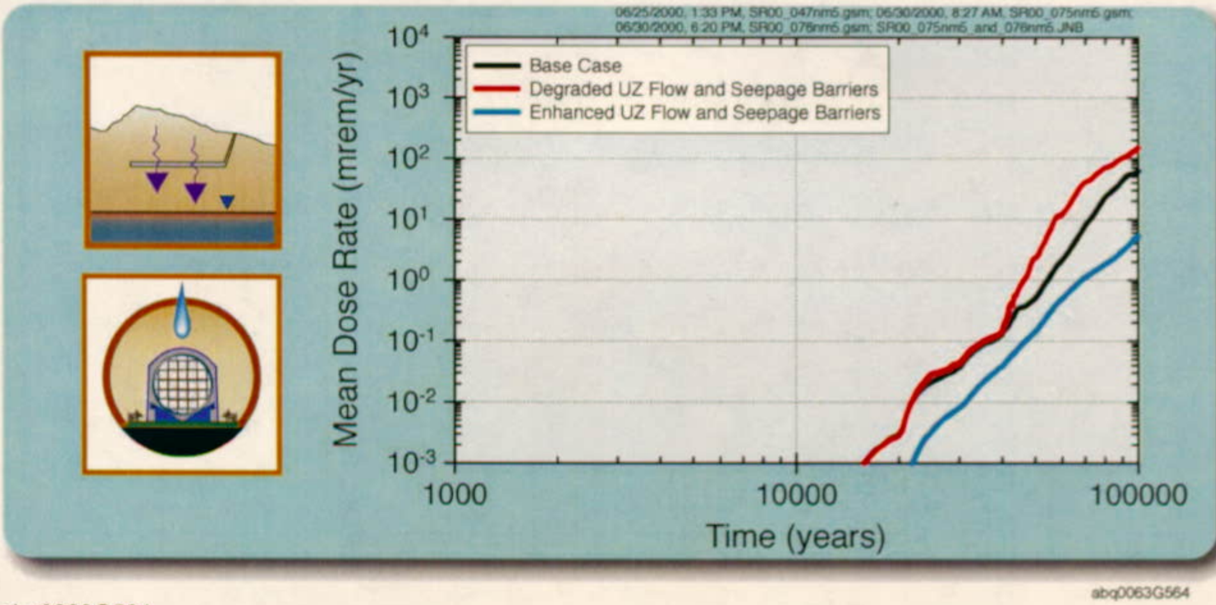
abq0063G619

Figure 5.2-27 Sensitivity Analysis of Mean Annual Dose for the Human Intrusion Base Case (an Intrusion Occurs at 100 Years after Potential Repository Closure) with the Infiltration Rate Fixed at the 95th-Percentile Value



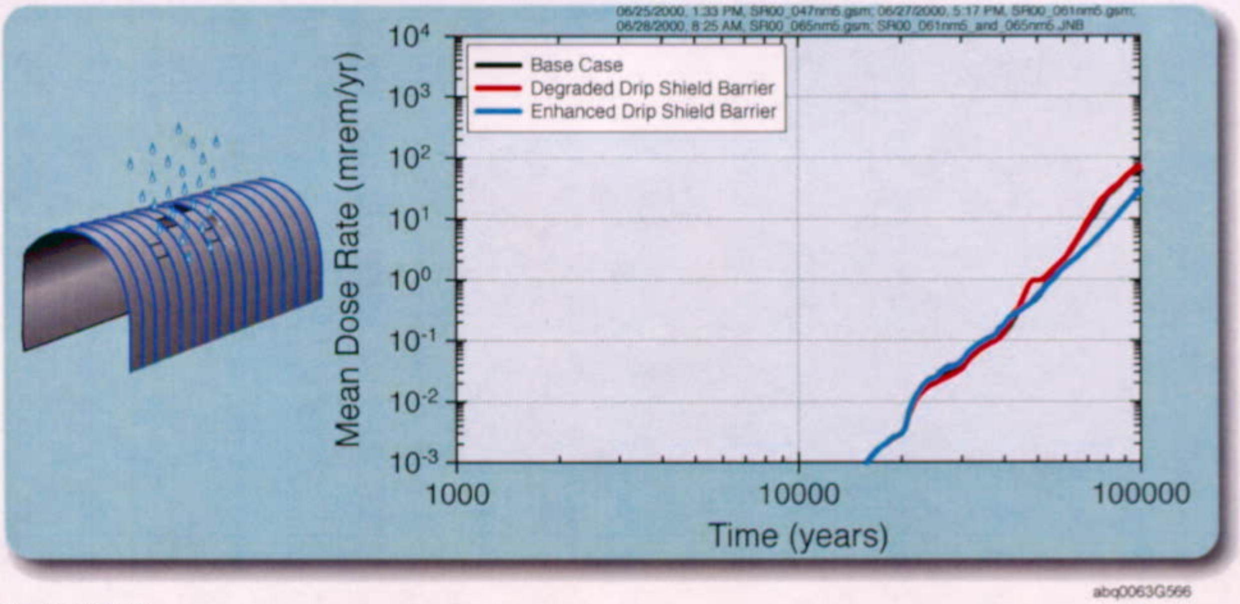
abq0063G563

Figure 5.3-1. Comparison of Mean Dose for Degraded and Enhanced Seepage Cases with the Base Case



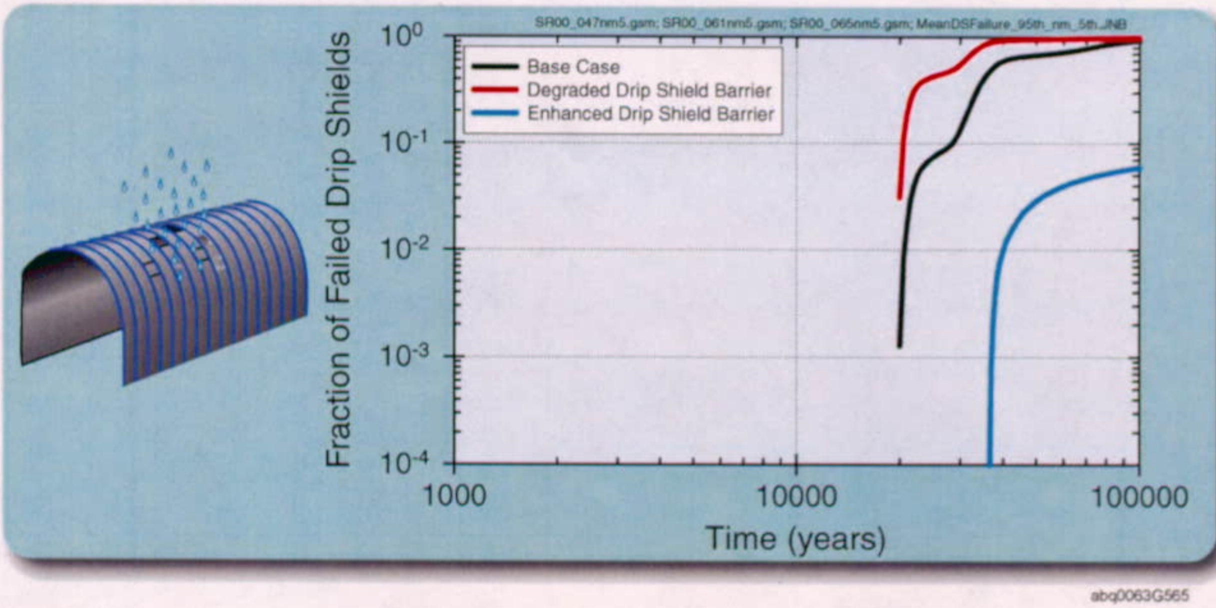
abq0063G564

Figure 5.3-2. Comparison of Mean Dose for Degraded and Enhanced Unsaturated Zone Flow and Seepage Cases with the Base Case



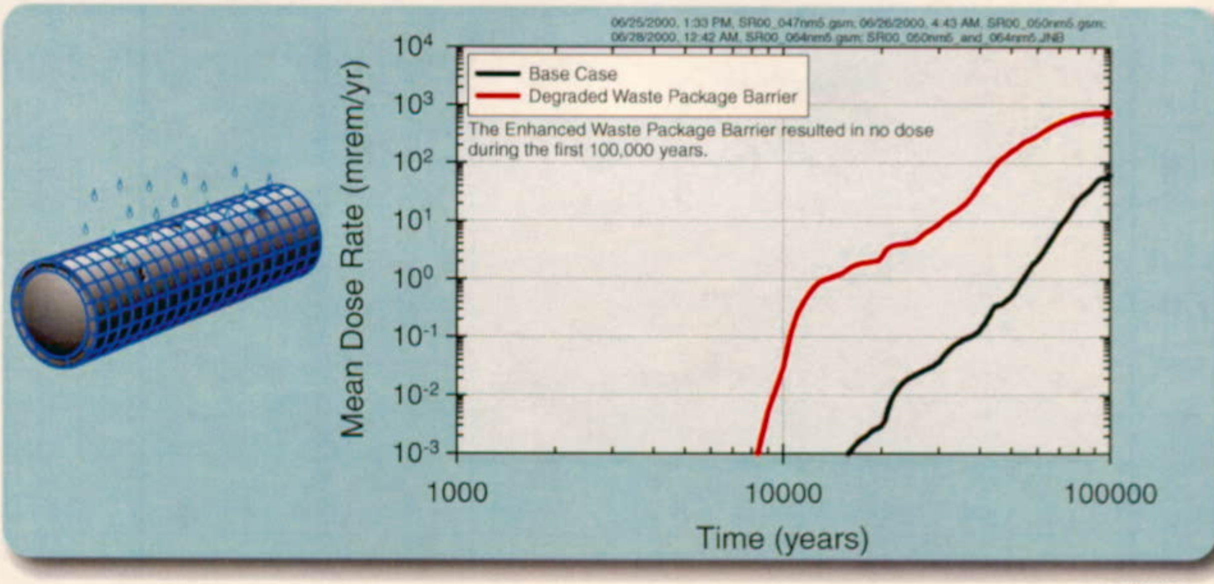
abq0063G566

Figure 5.3-3. Sensitivity of the Predicted Mean Dose Rate Profile to the Degraded and Enhanced Drip Shield Cases



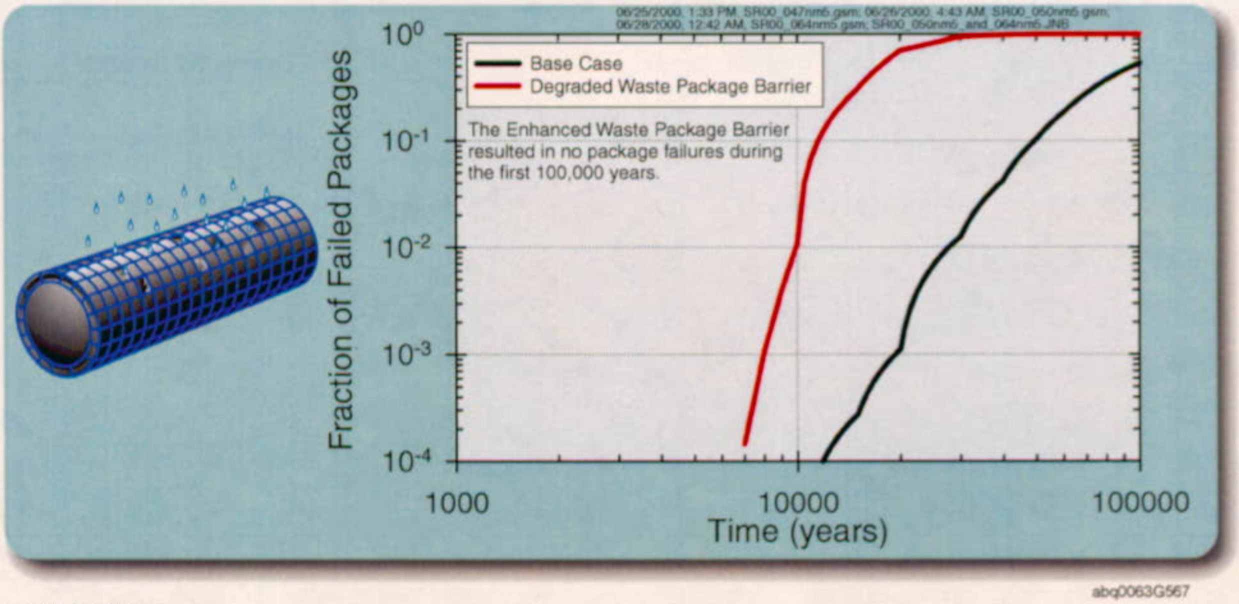
abq0063G565

Figure 5.3-4. Sensitivity of the Predicted Mean Drip Shield Failure Profile to the Degraded and Enhanced Drip Shield Cases



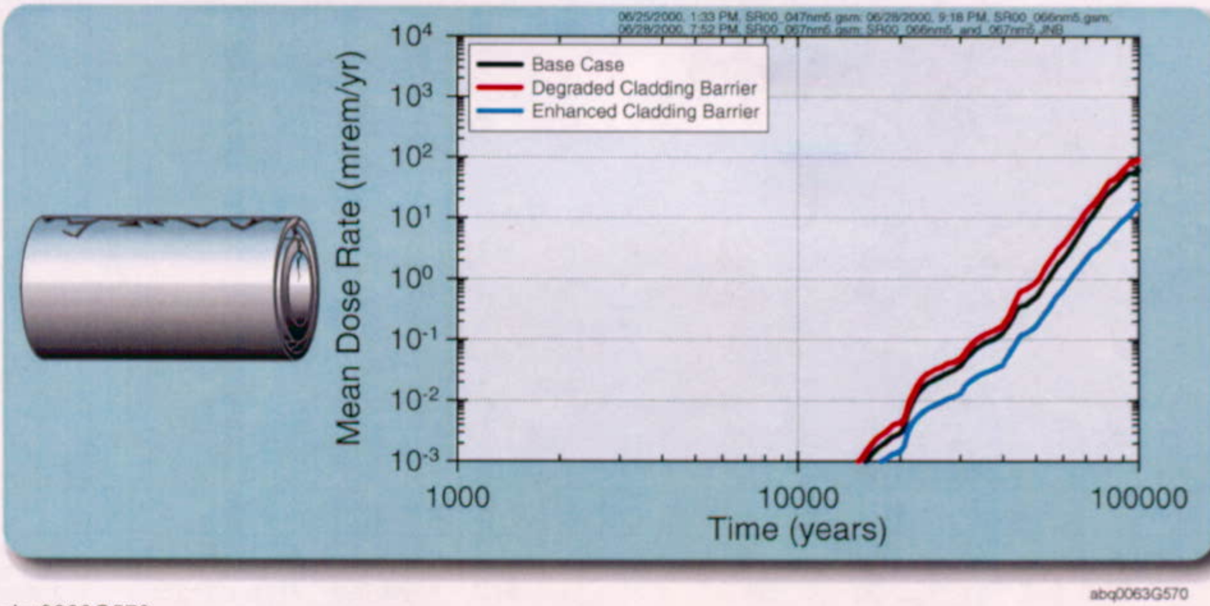
abq0063G568

Figure 5.3-5. Sensitivity of the Predicted Mean Dose Rate Profile to the Degraded and Enhanced Waste Package Cases



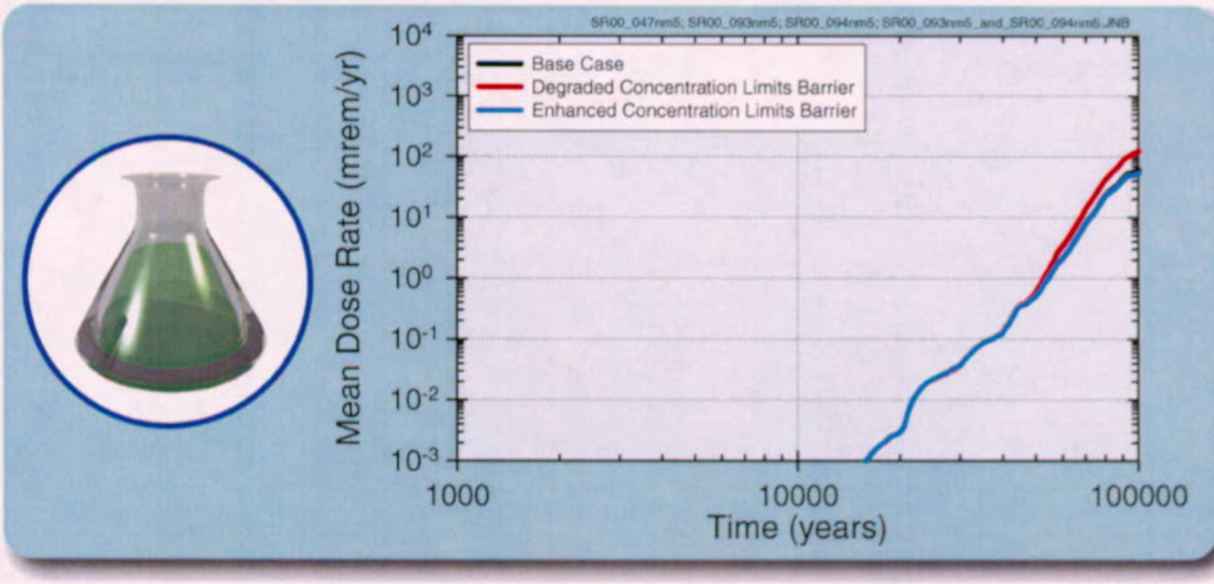
abq0063G567

Figure 5.3-6. Sensitivity of the Predicted Mean Waste Package Failure Profile to the Degraded and Enhanced Waste Package Cases



abq0063G570

Figure 5.3-7. Sensitivity of the Predicted Mean Dose Rate Profile to the Degraded and Enhanced Commercial Spent Nuclear Fuel Cladding Cases



abq0063G569

Figure 5.3-8. Comparison of Mean Dose for Degraded and Enhanced Concentration Limits with the Base Case

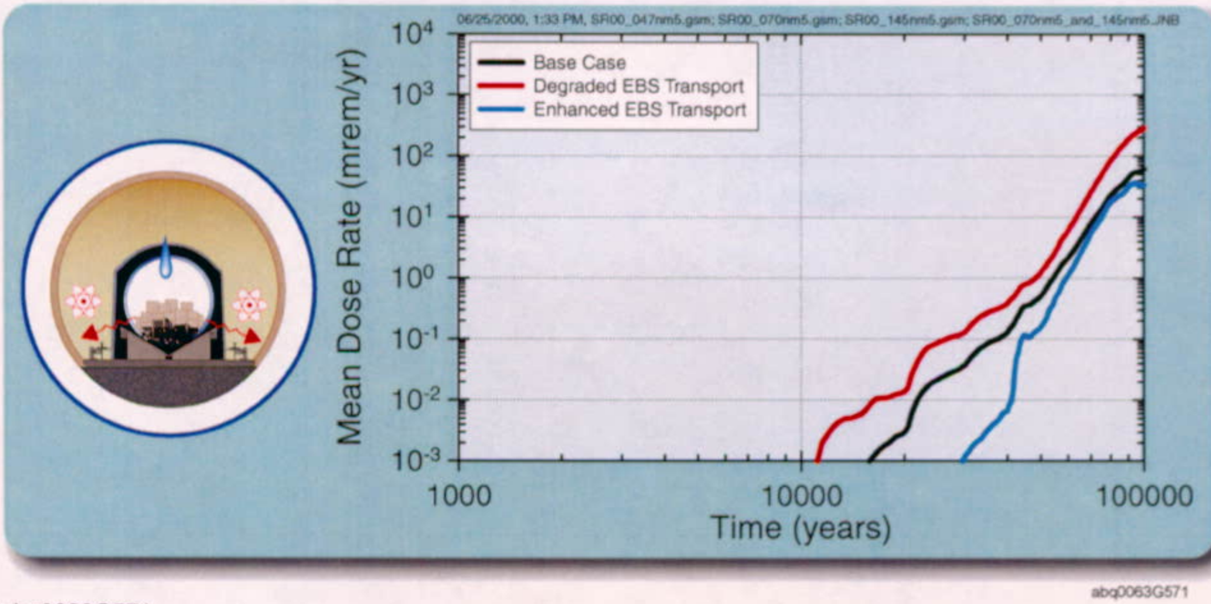


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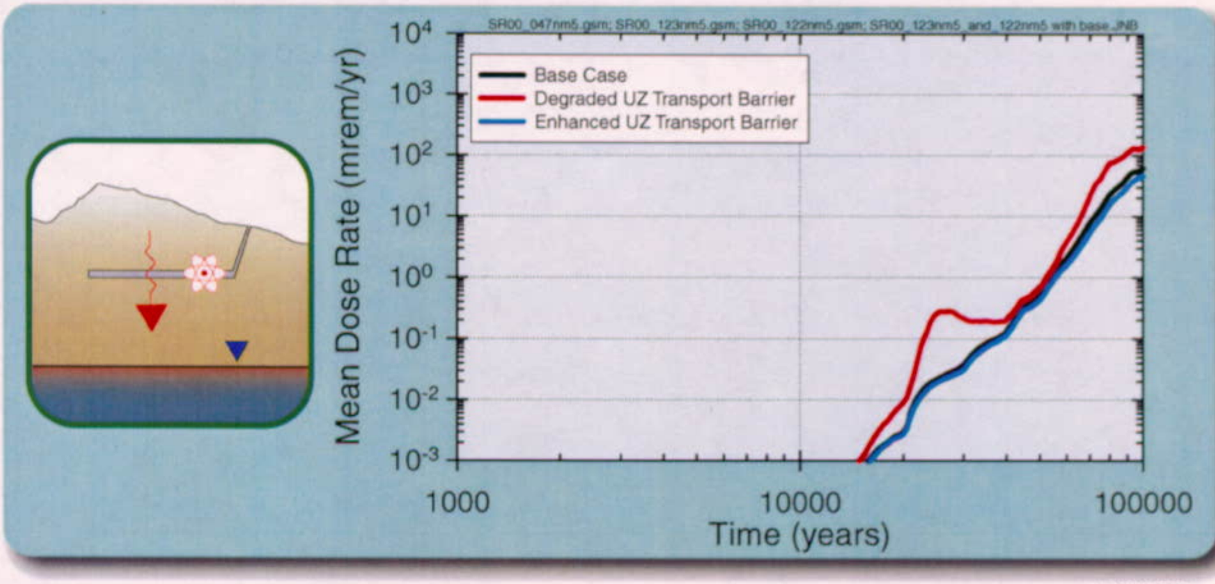
Figure 5.3-9. Comparison of Mean Dose for Degraded and Enhanced Engineered Barrier System Transport Cases with the Base Case



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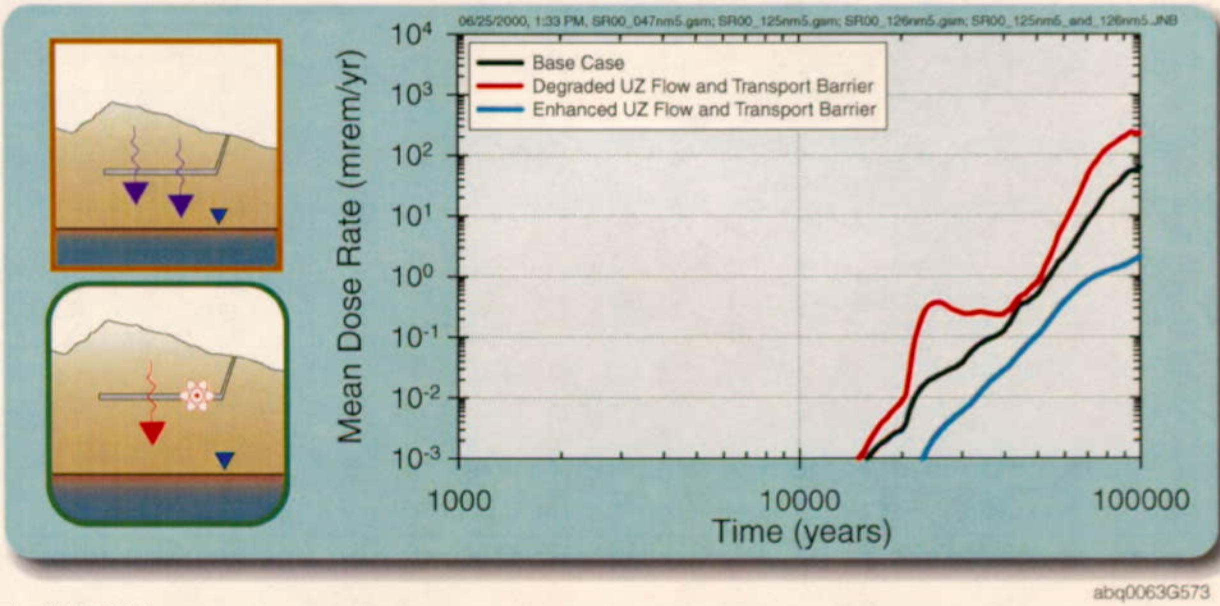
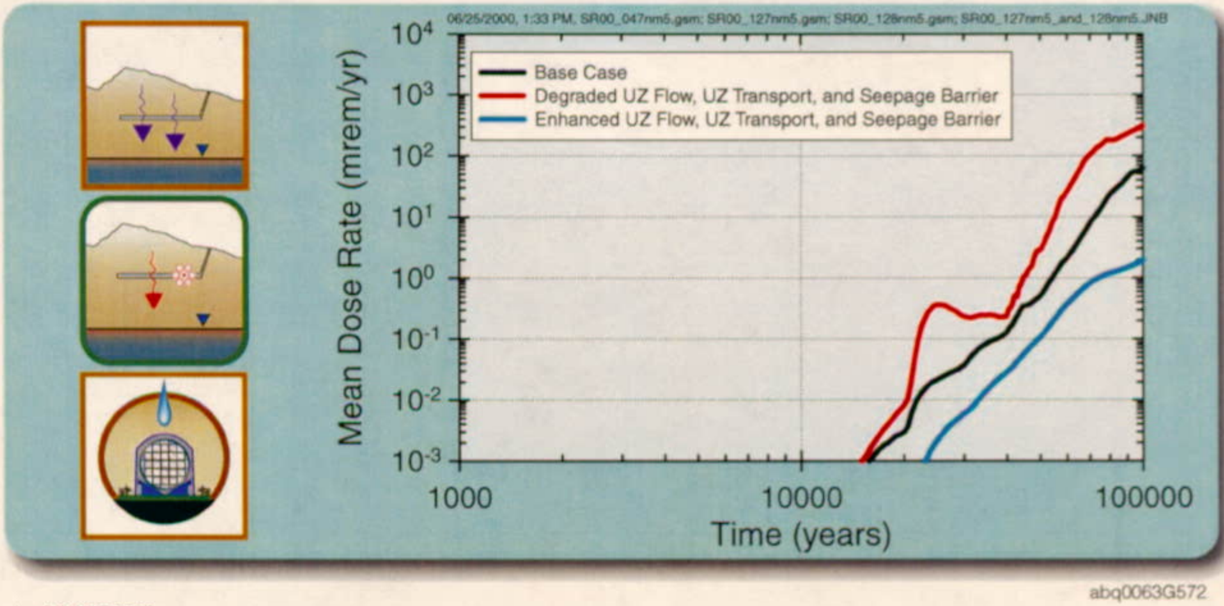
abq0063G574

Figure 5.3-10. Comparison of Mean Dose for Degraded and Enhanced Unsaturated Zone Transport Cases with the Base Case



abq0063G573

Figure 5.3-11. Comparison of Mean Dose for Degraded and Enhanced Unsaturated Zone Flow and Transport Cases with the Base Case



abq0063G572

Figure 5.3-12. Comparison of Mean Dose for Degraded and Enhanced Unsaturated Zone Flow, Transport, and Seepage Cases with the Base Case

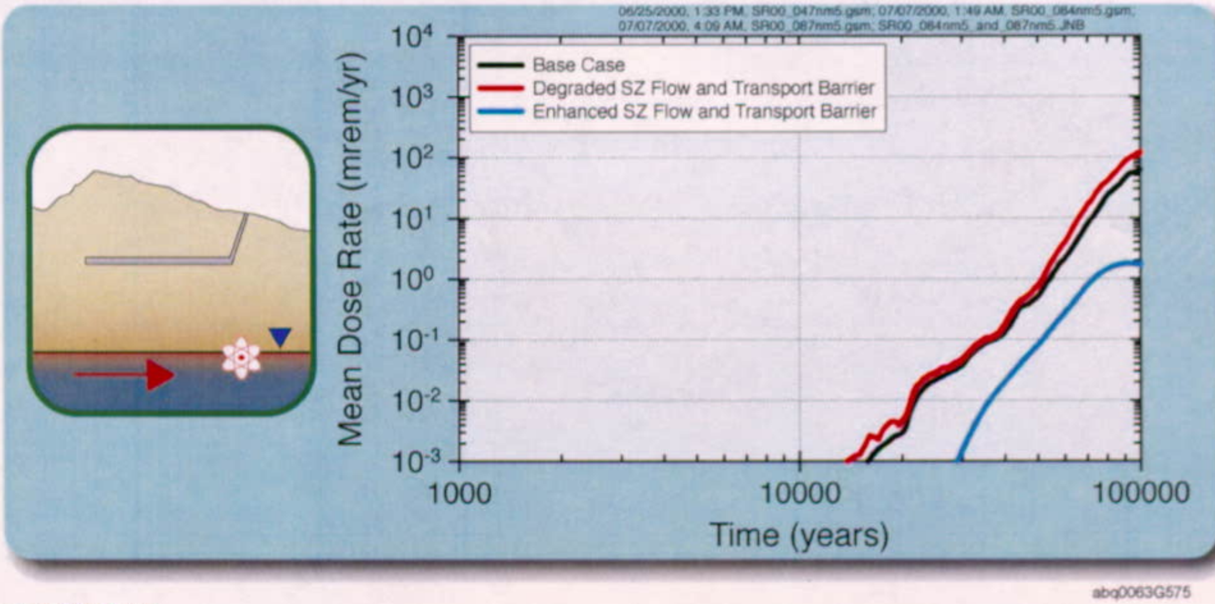


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Figure 5.3-13. Comparison of Mean Dose for Degraded and Enhanced Saturated Zone Flow and Transport Cases with the Base Case

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6. SUMMARY AND CONCLUSIONS

The TSPA-SR is the culmination of a body of scientific work that aims to evaluate the adequacy of the Yucca Mountain natural and engineered barriers in meeting postclosure public health and safety goals specified in applicable proposed regulations promulgated by the EPA, the NRC, and the DOE. The present document has presented how this scientific information has been integrated into a consistent picture of the overall repository system and has projected the future evolution of the potential repository system using each of the integrated component models.

This summary and conclusions section synthesizes the information presented in each of the preceding sections and attempts to provide another integration thread through the entire document. The aim of this presentation is not to repeat the details presented in the earlier sections of this document. Rather, it is to describe how the overall objectives have been addressed and to provide a roadmap for where particular issues have been addressed within the body of this document or in the Appendixes.

The discussion of summary and conclusions is broken into a summary of the overall system performance results (Section 6.1) which is followed by a discussion of the basis for these results (Section 6.2). The document concludes with a discussion of the intended use of these TSPA-SR results and conclusions in Section 6.3.

6.1 SUMMARY OF OVERALL SYSTEM PERFORMANCE RESULTS

There are three applicable proposed regulations that describe the requirements for performance assessments of the Yucca Mountain site: proposed 40 CFR Part 197 (64 FR 46976 [105065]), proposed 10 CFR Part 63 (64 FR 8640 [101680]), and proposed 10 CFR Part 963 (64 FR 67054 [124754]). These proposed regulations also describe the standards or performance objectives that the potential repository must meet to be of acceptable risk.

The requirements for a performance assessment specified in proposed 40 CFR Part 197 (64 FR 46976 [105065]) are:

> 197.20 Individual Protection Standard
> *The DOE must demonstrate, using performance assessment*, that there is a reasonable expectation that for 10,000 years following disposal the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microSv (15 mrem) from releases from the undisturbed Yucca Mountain disposal system. The DOE's analysis must include all potential pathways of radionuclide transport and exposure.

The requirements for a performance assessment specified in proposed 10 CFR Part 63 (64 FR 8640 [101680]) are:

> 63.113 Performance Objective For The Geologic Repository After Permanent Closure.

(c) The ability of the geologic repository to limit radiological exposures to those specified in 63.113(b) shall be demonstrated through a performance assessment that meets the requirements specified at 63.114, uses the reference biosphere and critical group specified at 63.115, and excludes the effects of human intrusion.

The requirements for a total system performance assessment specified in proposed 10 CFR Part 963 (64 FR 67054 [124754]) are:

963.15 Postclosure Suitability Determination.
DOE will apply the method and criteria described in Secs. 963.16 and 963.17 to evaluate the suitability of the Yucca Mountain site for the postclosure period. If DOE finds that the results of the total system performance assessments conducted under 963.16(a)(1) show that the Yucca Mountain site is likely to meet the applicable radiation protection standard, DOE may determine the site suitable for the postclosure period.

963.16 Postclosure Suitability Evaluation Method.
(a) DOE will evaluate postclosure suitability using the total system performance assessment method. DOE will conduct a total system performance assessment to evaluate the ability of the geologic repository to meet the applicable radiation protection standard.

Given these requirements for a performance assessment, it useful to define this term in the context used in each of the above regulations.

The definition of performance assessment as used in proposed 40 CFR Part 197 (64 FR 46976 [105065]) is:

Performance assessment means an analysis that:
(1) Identifies the processes, events, and sequences of processes and events (except human intrusion), and their probabilities of occurring over 10,000 years after disposal, that might, affect the Yucca Mountain disposal system;
(2) Examines the effects of those processes, events, and sequences of processes and events upon the performance of the disposal system; and
(3) Estimates the annual committed effective dose equivalent received by the reasonably maximally exposed individual, including the associated uncertainties, as a result of releases caused by all significant processes, events, and sequences of processes and events.

The definition of performance assessment as used in proposed 10 CFR Part 63 (64 FR 8640 [101680]) is:

Performance assessment means a probabilistic analysis that:
(1) Identifies the features, events and processes that might affect the performance of the geologic repository; and
(2) Examines the effects of such features, events and processes on the performance of the geologic repository; and

(3) Estimates the expected annual dose to the average member of the critical group as a result of releases from the geologic repository.

The definition of total system performance assessment as used in proposed 10 CFR Part 963 (64 FR 67054 [124754]) is:

> *Total System performance assessment* means a probabilistic analysis that is used to:
> (1) Identify the features, events and processes that might affect the performance of the geologic repository;
> (2) Examines the effects of such features, events and processes on the performance of the geologic repository; and
> (3) Estimates the expected annual dose to the receptor as a result of releases from the geologic repository.

This document, along with the supporting references cited in this document, contains all of the elements of the performance assessment outlined in the above requirements. Specifically, the relevant FEPs that might affect the performance have been presented in the appropriate subsections of Section 3. The details of the FEPs screening process and results are contained in Appendix B which cites the various FEPs screening AMRs which contain the technical basis for the screening arguments. The models developed to analyze the effects of the FEPs are presented in summary form in the various subsections of Section 3, while the effects of these FEPs are presented in the results described in Sections 4 and 5. Finally, the expected annual dose, as well as the uncertainty in the expected annual dose and the significance of the individual FEPs to the expected annual dose are presented in Sections 4 and 5.

Four post-closure performance objectives have been specified in the above proposed regulatory requirements. These four objectives are:

- Individual protection standard
- Human intrusion standard
- Groundwater protection standard
- Peak dose (required in proposed 40 CFR 197.30 [64 FR 46976 [105065]]).

The following sections summarize the TSPA-SR results with respect to these four post-closure performance objectives. Additional details of how these results have been generated are contained in Sections 4.1 through 4.3 (for individual protection), Section 4.4 (for human intrusion), Section 4.1.5 (for groundwater protection), and Section 4.1.3 (for peak dose). It bears noting that although the individual protection standard is the only post-closure performance objective that explicitly requires a performance assessment, the analyses to evaluate the other performance objectives have used the same methodology and models as used in the individual protection analysis. The differences are noted in the appropriate section of Chapter 4.

6.1.1 Summary of Individual Protection Performance Results

The requirements for individual protection performance as specified in proposed 40 CFR Part 197 (64 FR 46976 [105065]) are:

> 197.20 Individual-Protection Standard.
> The DOE must demonstrate, using performance assessment, that there is a reasonable expectation that for 10,000 years following disposal the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microSv (15 mrem) from releases from the undisturbed Yucca Mountain disposal system. The DOE's analysis must include all potential pathways of radionuclide transport and exposure.

The requirements for individual protection performance as specified in proposed 10 CFR Part 63 (64 FR 8640 [101680]) are:

> 63.113 Performance Objective For The Geologic Repository After Permanent Closure.
> (b) The EBS shall be designed so that, working in combination with natural barriers, the expected annual dose to the average member of the critical group shall not exceed 0.25 mSv (25 mrem) TEDE at any time during the first 10,000 years after permanent closure, as a result of radioactive materials released from the geologic repository.

The requirements for individual protection performance as specified in proposed 10 CFR Part 963 (64 FR 67054 [124754]) are:

> 963.16 Postclosure Suitability Evaluation Method.
> (1)DOE will conduct a [TSPA] to evaluate the ability of the geologic repository to limit radiological exposures in the case where there is no human intrusion into the repository. DOE will model the performance of the geologic repository at the Yucca Mountain site using the method described in 963.16(b) and the criteria in Sec 963.17, excluding the criterion in 963.17(b)(4). DOE will consider the performance of the system in terms of the criteria to evaluate whether the geologic repository is likely to comply with the applicable radiation protection standard.

For the purpose of presenting the individual protection performance results, the analyses have been subdivided into a nominal performance scenario class and a volcanic event scenario class. Projections of the individual dose for the nominal scenario class are presented in Figure 6.1-1. Projections of the individual dose for the volcanic event scenario class are presented in Figure 6.1-2. Figure 6.1-3 combines these results to yield the projected total system dose to the individual.

Several points are worth summarizing on these projections:

- These projections have included the uncertainty in the dose attributed to the quantified uncertainty discussed in the process models, abstraction models and their included parameters presented in Section 3. As a result, a distribution of potential doses has been developed reflecting this uncertainty.

- Additional unquantified uncertainty exists, the impacts of which have not been captured in the results presented. These unquantified uncertainties reflect in large part the conservative assumptions included in the analyses to increase the defensibility in particularly complex processes. Appendix F summarizes the most significance of these conservative assumptions included in the analyses. The result of this conservatism is to over-predict the possible performance.

- Projections are made beyond the 10,000-year regulatory time period specified in proposed 40 CFR Part 197 (64 FR 46976 [105065]) and proposed 10 CFR Part 63 (64 FR 8640 [101680]). These projections are important to evaluate the robustness of the system response and the contribution of both natural and engineered barriers to the overall system performance during the time period when the containment of the engineered barrier is being degraded. In addition to these post-10,000 year projections being performed to assure that no dramatic degradation of the performance occurs after the compliance period, they will also be used in the EIS to consider the effects of the peak dose.

- Projections are made for the dose to the individual residing 20 kilometers downgradient from the potential repository, in the vicinity of Lathrop Wells.

- These projections are applicable to both the average member of the critical group (using the definition of proposed 10 CFR Part 63 [64 FR 8640 [101680]]) and the reasonably maximally exposed individual (using the definition of proposed 40 CFR Part 197 [64 FR 46976 [105065]]). It is important to recognize that both of these individuals reside within a group of individuals likely to be most exposed to the risks associated with the long-term performance of the potential repository (given that the group of individuals is assumed to reside over the plume of contaminated groundwater). The exact characteristics of this individual (whether s/he is the "average member" or the "reasonably maximally exposed individual") are similar. In both instances the individual: (a) has a fraction of their diet based on the consumption of locally grown produce, milk, and meat; (b) lives in the vicinity of Lathrop Wells; (c) has a lifestyle consistent with the existing population of Amargosa Valley; and (d) drinks 2 liters of water per day derived from the contaminated groundwater.

- The projected doses for the volcanic event scenario class reflect probability-weighted doses as required in proposed 10 CFR Part 63 (64 FR 8640 [101680]); that is the probability of the event is multiplied by the dose consequence to yield the dose risk. This allows combining the nominal and volcanic event scenario classes to yield the total system dose response of Figure 6.1-3.

Based on the above, it is reasonable to conclude that the potential Yucca Mountain repository system is likely to meet the individual protection requirements of both proposed 10 CFR Part 63 (64 FR 8640 [101680]) and proposed 40 CFR Part 197 (64 FR 46976 [105065]).

6.1.2 Summary of Human Intrusion Performance Results

The requirements for human intrusion performance as specified in proposed 40 CFR Part 197 (64 FR 46976 [105065]) are:

> 197.25 Human Intrusion Standard.
> Alternative 1:
> The DOE must demonstrate that there is a reasonable expectation that for 10,000 years following disposal the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microSv (15 mrem) as a result of a human intrusion. The DOE analysis of human intrusion must include all potential environmental pathways of radionuclide transport and exposure.
>
> Alternative 2:
> The DOE must determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion (see 197.26) could occur without recognition by the drillers. The DOE must:
> (a) Demonstrate that there is a reasonable expectation that the reasonably maximally exposed individual receives no more than an annual committed effective dose equivalent of 150 microSv (15 mrem) as a result of a human intrusion, if complete waste package penetration can occur at or before 10,000 years after disposal. The analysis must include all potential environmental pathways of radionuclide transport and exposure; and
> (b) Include the results of the analysis and its bases in the environmental impact statement for Yucca Mountain as an indicator of long-term disposal system performance, if the intrusion cannot occur before 10,000 years after disposal.

The requirements for human intrusion performance as specified in proposed 10 CFR Part 63 (64 FR 8640 [101680]) are:

> 63.113 Performance Objective for the Geologic Repository After Permanent Closure.
> (d) The ability of the geologic repository to limit radiological exposures to those specified in §63.113(b), in the event of limited human intrusion into the EBS, shall be demonstrated through a separate performance assessment that meets the requirements specified at 63.114 and uses the reference biosphere and critical group specified at 63.115. For the assessment required by this paragraph, it shall be assumed that the human intrusion occurs 100 years after permanent closure and takes the form of a drilling event that results in a single, nearly vertical borehole that penetrates a waste package, extends to the SZ, and is not adequately sealed.

The requirements for human intrusion performance as specified in proposed 10 CFR Part 963 (64 FR 67054 [124754]) are:

> 963.16 Postclosure Suitability Evaluation Method.
> (a)(2) Consistent with applicable NRC regulations regarding a stylized human intrusion case, DOE will conduct a [TSPA] to evaluate the ability of the geologic

repository to limit radiological exposures in a stylized limited human intrusion case. DOE will model the performance of the geologic repository at the Yucca Mountain site using the method described in 963.16(b) and the criteria in Sec 963.17. DOE will consider the performance of the system in terms of the criteria to evaluate whether the geologic repository is likely to comply with the applicable radiation protection standard. The human intrusion evaluation under this paragraph will be separate from the evaluation conducted under 963.16(a)(1).

Human intrusion analyses have been conducted by assuming a borehole penetrates the engineered barriers (the drip shield, waste package, cladding, and invert) and provides a pathway from the surface to the repository and from the repository to the water table. Analyses have been conducted assuming the human intrusion occurs at 100 years (per proposed 10 CFR Part 63 [64 FR 8640 [101680]]) or 10,000 years (which would be required in the EIS even if the final EPA rule uses Alternative 2 of the proposed 40 CFR Part 197.25 (64 FR 46976 [105065]) and the DOE is able to demonstrate that the intrusion could not occur before 10,000 years). The results for these analyses are presented in Figures 6.1-4 and 6.1-5, respectively.

Several points are worth summarizing regarding the projections of the potential consequences associated with the stylized human intrusion scenario:

- It is highly unlikely that a borehole would be drilled from the surface of Yucca Mountain and that the driller would not detect the presence of the drift (due to loss of drilling fluid), or the drip shield, or the waste package (due to the difficulty in drilling through these metals). However, for the purposes of the analyses performed for the 100-year intrusion event, all of these considerations have been ignored.

- Should the final regulation appear more like alternative 2 of the proposed EPA standard, credit for the robustness of the engineered barriers during the intrusion event may be considered. In this case, the 10,000-year intrusion event would still be germane and included in the environmental impact statement.

- The uncertainty in the projected dose from the event is controlled by the uncertainty in the flow rates through the borehole, the concentration of the radionuclides in the intruded waste package, and the uncertainty in the SZ flow and transport characteristics.

Based on the above, it is reasonable to conclude that the potential Yucca Mountain repository system is likely to meet the human intrusion requirements of both proposed 10 CFR Part 63 (64 FR 8640 [101680]) and proposed 40 CFR Part 197 (64 FR 46976 [105065]).

6.1.3 Summary of Groundwater Protection Performance Results

The requirements for groundwater protection performance as specified in proposed 40 CFR Part 197 (64 FR 46976 [105065]) are:

197.35 What Standards Must DOE Meet?
In its license application to NRC, DOE must provide a reasonable expectation that, for 10,000 years of undisturbed performance after disposal, release of radionuclides from radioactive material in the Yucca Mountain disposal system

> will not cause the level of radioactivity in the representative volume of groundwater at the point of compliance to exceed the limits in [the table] as follows:

Radionuclide or type of radiation emitted	Limit	Is natural background included?
Combined ^{226}Ra and ^{228}Ra	5 picocuries per liter	Yes
Gross alpha activity (including ^{226}Ra, but excluding radon and uranium)	15 picocuries per liter	Yes
Combined beta and photon emitting radionuclides	40 microsieverts (4mrem) per year to the whole body or any organ	No

The results of the groundwater protection analyses are illustrated in Figure 6.1-6 and 6.1-7 for the concentration and dose performance measures, respectively. Figure 6.1-6 illustrates the combined ^{226}Ra and ^{228}Ra concentrations in the representative volume of groundwater, as well as the gross alpha activity concentration (including ^{226}Ra, but excluding radon and uranium). Figure 6.1-7 shows the dose associated with the beta and photon emitting radionuclides (^{129}I, ^{99}Tc, and ^{14}C). The critical organ for each of these three radionuclides is the thyroid (for ^{129}I), the gastrointestinal tract (for ^{99}Tc), and fat (for ^{14}C). Several summary observations are possible from these analyses:

- Although the regulatory time period for groundwater protection is 10,000 years, the analyses have been extended to 100,000 years to illustrate the long-term behavior of the system. As noted in the individual protection analyses, these longer term projections are useful to assure that no significant degradation of the performance occurs after the 10,000 year time period of regulatory concern and to provide input to longer term assessments required in the EIS.

- The groundwater protection analyses assumed a representative water volume of 1,285 acre-ft/yr. centered on the highest concentration in the plume of contamination within the freshwater aquifer. In this analysis, all radionuclides that reach a distance of 20 km (12 miles) from the potential repository in any given annual period are contained in 1,285 acre-ft of water to determine the concentration. Taking all radionuclides in this manner produces the highest estimate of concentration that is possible for the specified volume of water. It is not necessary to specify an exact location or dimensions for the representative volume, which could be different in each Monte Carlo realization with this method.

- This projection considers undisturbed performance (i.e., performance not disturbed by the potential consequences associated with low-probability disruptive events) such as volcanism.

- The natural background concentrations are not illustrated on these plots. As noted in *Radioactivity in FY 1998 Groundwater Samples from Wells and Springs Near Yucca*

Mountain (CRWMS M&O 1999 [150420], pp. 8 to 9), the gross alpha activity background concentration ranges from –0.2 ± 0.5 to 2.7 ± 3.0 pCi/L, with the well closest to the 20-kilometer compliance location having a concentration of 0.4 ± 0.7 pCi/L. Concentrations of ^{226}Ra and ^{228}Ra were not reported because the gross alpha activity was below 5 pCi/L. However, another source reports ^{226}Ra concentrations in the well closest to the compliance point of 0.04 pCi/L and ^{228}Ra concentrations less than 1 pCi/L (DTN: GS971000012847.004 [149980]). The measurement errors are not reported in this reference. Measurements in the same reference at other wells and springs in the vicinity of Yucca Mountain and Amargosa Valley range from 0.03 to 0.5 pCi/L for ^{226}Ra and from less than 1 to 1.1 pCi/L for ^{228}Ra.

Based on the above, it is reasonable to conclude that the potential Yucca Mountain repository system is likely to meet the groundwater protection requirements of proposed 40 CFR Part 197 (64 FR 46976 [105065]).

6.1.4 Summary of Peak Dose Performance Results

The requirements for peak dose performance as specified in proposed 40 CFR Part 197 (64 FR 46976 [105065]) are:

> 197.30 What Other Projections Must Be Made by DOE?
> To complement the results of 197.20, DOE must calculate the peak dose of the reasonably maximally exposed individual that would occur after 10,000 years following disposal but within the period of geologic stability. While no regulatory standard applies to the results of this analysis, DOE must include the results and their bases in the environmental impact statement for Yucca Mountain as an indicator of long-term disposal system performance.

The results of the peak dose performance assessments are illustrated in Figure 6.1-8 for three different representations of the models used to project the peak dose. Several summary observations are possible from the following results:

- The time period of geologic stability is considered to be 1,000,000 years as suggested by the National Academy of Sciences.
- The peak dose occurs after the engineered barriers have been degraded sufficiently to allow advective flux of groundwater into all of the waste packages that are contacted by seepage.
- The expected value of the peak dose is a function of the degree of conservatism incorporated in the models and analyses used to produce the peak dose estimate. Because the base case models used in the development of the nominal performance projections were designed to be reasonably conservative to maximize their defensibility during the 10,000-year compliance period, they are less appropriate for projections of the peak dose. More appropriate representations would include considerations of the long-term (post-10,000-year) climate states and the long term effects of secondary phases.

- Depending on the representation considered, the peak dose varies from 460 mrem/yr for the case of extending the conservative models developed for the 10,000-year compliance analyses to the time of the peak dose, to 120 mrem/yr for the case of extending all the conservative models except the post-10,000 year climate model (described in Section 3.2.5) and the long-term secondary phase solubility limit model (described in Section 3.5.), to 30 mrem/yr for the case of extending all the conservative models except the long-term secondary phase solubility model.

- The variance in the peak dose magnitude is relatively small (a few orders of magnitude), because at that time all of the uncertainty associated with engineered barrier performance and the travel time in the natural barrier are insignificant to the magnitude of the peak dose.

As noted in proposed 40 CFR 197.30 (64 FR 46976 [105065]), no regulatory standard applies to the results of the peak dose analyses. They are provided to support the development of the environmental impact statement. Although these results do provide insights into the possible long term performance of a repository at Yucca Mountain, they should not be interpreted as accurate predictions of the likely performance over these time periods due to the large uncertainties and conservative approximations included in the models that were designed for assessing the 10,000-year compliance performance.

6.2 SUMMARY OF TECHNICAL BASIS OF OVERALL SYSTEM PERFORMANCE RESULTS

The projections of total system performance summarized in the previous section must be interpreted in light of their technical foundation. The technical basis for the TSPA analyses are contained within a family of AMRs that have been summarized in nine PMRs. The integration of the analyses and models in the context of the TSPA-SR model is presented in detail in the TSPA-SR Model Document (CRWMS M&O 2000 [148384]). The purpose of this section is to provide the reader with a series of roadmaps that depict where the technical basis is presented. Additional roadmaps are presented in the appendices to this document, in particular Appendix E which present the information flow used to develop the TSPA-SR model and Appendix F which summarizes the major assumptions and conservatisms used in the TSPA-SR model.

The summary of the technical basis for a complex system, such as the postclosure performance model of the potential Yucca Mountain repository system, is a daunting task. In order to provide some rational logic to the presentation, the discussion is broken into the following major topics:

- How the TSPA-SR has provided an integrated and traceable analysis using the family of over one hundred AMRs which have been summarized in the nine PMRs.

- How the TSPA-SR has addressed the uncertainty and variability in the component models and evaluated the significance of this uncertainty in the projection of overall performance.

- How the TSPA-SR has addressed both technical and process recommendations made during the generation of earlier TSPAs and by reviewers of earlier TSPA analyses, most notably the TSPA-VA completed in 1998.

- How the TSPA-SR has addressed the goals and objectives outlined in proposed regulatory requirements (notably proposed 40 CFR Part 197 [64 FR 46976 [105065]], proposed 10 CFR Part 63 [64 FR 8640 [101680]], and proposed 10 CFR Part 963 [64 FR 67054 [124754]]) along with the NRC's Acceptance Criteria noted in their Issue Resolution Status Report on Total System Performance Assessment and Integration (NRC 2000 [149372]).

- How the TSPA-SR may be used to address the regulatory requirements and other supporting information that may be required of decision makers.

The first three of these items are addressed in this section. The last two items are addressed in Section 6.3. With this information, combined with the information presented in the technical discussions in the previous chapters, the interested reader, whether a policy maker, decision maker, regulatory reviewer, or member of the public, can make an informed decision on the adequacy of the analysis for the intended purpose of evaluating the suitability of the potential Yucca Mountain repository system.

6.2.1 Summary of Traceability and Transparency of the Integrated TSPA-SR Analyses

An overall objective of any integrated performance assessment, but in particular total system performance assessments of potential nuclear waste repositories, is to provide a "transparent and traceable" analysis that allows the reader the opportunity to understand the basic assumptions and their scientific basis in such a way that he/she may understand and test the accuracy and reproducibility of the conclusions. Although no common definitions of these terms exist, the Nuclear Energy Agency has defined transparency as a document written in such a way that the reader can gain a clear understanding of what has been done, what the results are, and why the results are as they are (Nuclear Energy Agency 1998 [111738]). The Nuclear Energy Agency has defined traceability as an assessment that provides a complete record of the decisions and assumptions made and the models and data used to arrive at a given result.

Throughout this report, the underlying data, assumptions, models, and analyses have been discussed with appropriate conceptual drawings and integration graphics to illustrate the role of the component model, the technical basis of the component model, and the information flow from or to each component model. In addition, interim results have been presented both at the component level in Chapter 3 and the subsystem level in Chapter 4 to illustrate how information (in terms of mass, water, energy, activity) flows from one component of the system to the next in the integrated total system model. Finally, Appendix E presents the hierarchy of all analysis model reports that support the final information feed to the TSPA-SR model.

In the presentation of the individual component models that form the technical foundation of the TSPA-SR model, we have lumped the processes into "process model factors." This subdivision allows a convenient way of illustrating not only how and where that component fits into the total system representation, but also provides a traceable roadmap for summarizing the inclusion or exclusion of relevant features, events, and processes (as documented in Appendix B). In addition, these process model factors allow for a convenient means of lumping the key input parameters for the TSPA-SR model and the supporting documentation where these parameters are discussed in more detail.

Table 6.2-1 summarizes the source (i.e., the relevant AMR) of the technical basis for each of the process model factors or model components used in the TSPA-SR model. The process model report which summarizes and synthesizes the technical defensibility of the analysis model reports is also indicated. In addition, this table provides a roadmap to figures in Section 3 and 4 where the intermediate performance results and key parameters affecting system performance are presented.

The defensibility of the analyses and models which support the TSPA-SR model is contained in the relevant AMRs and PMRs. It is the analysis model reports and process model reports which provide the fundamental scientific underpinning, and the associated assumptions and conservatisms necessary for a defensible, yet reasonably cautious analysis of expected performance.

It is beyond the scope of this document to summarize the depth and breadth of the information contained in the analysis model reports and process model reports that form the basis for the TSPA-SR. Suffice it to say that the individual models are based on appropriate site-specific information, analog data, and relevant literature data sources that have been integrated by the principal scientific investigators to provide a reasonable and defensible characterization of each individual process relevant to postclosure performance. As discussed in the following section, quantifiable uncertainty in the individual component model was included as appropriate. Where the individual process model was subject to significance complexity or the available information did not allow a definitive conclusion regarding the most reasonable representation, the analyst or modeler chose to apply some conservatism to the individual model. Areas where conservative representations were employed and the basis for that conservatism are enumerated in Appendix F.

It is these AMRs which provide the fundamental scientific underpinning, associated assumptions, and conservatisms necessary for a defensible, yet reasonably cautious analysis of expected performance.

In addition to the analysis model reports providing a traceable chain of references for the defensibility of the scientific bases for the TSPA-SR, they also provide a hierarchy of data tracking numbers. Appendix E summarizes the sources and hierarchy of data sets used as input to the TSPA-SR model. Additional details of the data sets used as input are contained in the TSPA-SR Model Report (CRWMS M&O 2000 [148384]). The quality status of each data set used as input to the TSPA-SR model can be ascertained by tracing the data set and all its predecessors using the Document Input Reference System database. This capability allows the DOE and NRC to track the status of all data sets used in the development of the postclosure safety case.

Table 6.2-1. Summary of Analysis Model Reports, Process Model Reports, and Figures Illustrating Key Input Parameters to Total System Performance Assessment-Site Recommendation

Key Attributes of System	Factor	Analysis Model Report	Process Model Report	Figure Illustrating Key Input Parameters or Intermediate Performance Results
Limiting Water Contacting Waste Package	Climate	*Future Climate Analysis* (USGS 2000 [136368])	UZ[a]	
	Net Infiltration	*Analysis of Infiltration Uncertainty* (CRWMS M&O 2000 [143244])	UZ[a]	3.2-7
	UZ Flow	*Abstraction of Flow Fields for RIP* (CRWMS M&O 2000 [123913])	UZ[a]	3.2-8
	Coupled Effects on UZ Flow	*Drift Scale Coupled Processes (DST and THC Seepage) Models* (CRWMS M&O 2000 [141389])	UZ[a]	
	Seepage into Emplacement Drifts	*Abstraction of Drift Seepage* (CRWMS M&O 2000 [142004])	UZ[a]	3.2-15
		Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux (CRWMS M&O 2000 [152204])	NFE[b] EBS[c]	
	Coupled Effects on Seepage	*Abstraction of Drift Seepage* (CRWMS M&O 2000 [142004])	UZ[a]	3.2-15
Long Waste Package Lifetime	In-Drift Physical and Chemical Environments	*Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	NFE[b] EBS[c]	3.3-9 3.3-10
		In-Drift Precipitates/Salts Analysis. (CRWMS M&O 2000 [127818])	EBS[c]	
	In-Drift Moisture Distribution	*EBS Radionuclide Transport Abstraction Model* (CRWMS M&O 2000 [129284])	EBS[c]	
	Drip Shield Degradation and Performance	*Analysis of Mechanisms for Early Waste Package Failure* (CRWMS M&O 2000 [147359])	WP[d]	
		Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis (CRWMS M&O 2000 [147641]) *WAPDEG Analysis of Waste Package and Drip Shield Degradation* (CRWMS M&O 2000 [146427])	WP[d]	3.4-11 3.4-16 3.4-17 4.1-7

Table 6.2-1. Summary of Analysis Model Reports, Process Model Reports, and Figures Illustrating Key Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Factor	Analysis Model Report	Process Model Report	Figure Illustrating Key Input Parameters or Intermediate Performance Results
Long Waste Package Lifetime (Continued)	Waste Package Degradation and Performance	Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier. (CRWMS M&O 2000 [146460])	WP[d]	
		Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis. (CRWMS M&O 2000 [147641])	WP[d]	3.4-10 3.4-12 3.4-13
		WAPDEG Analysis of Waste Package and Drip Shield Degradation (CRWMS M&O 2000 [146427])	WP[d]	3.4-16 4.1-8
		Aging and Phase Stability of Waste Package Outer Barrier (CRWMS M&O 2000 [147639])	WP[d]	
		Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier (CRWMS M&O 2000 [146460])	WP[d]	
		Abstraction of Models for Pitting and Crevice Corrosion of Drip Shield and Waste Package Outer Barrier (CRWMS M&O 2000 [147648])	WP[d]	
		Abstraction of Models for Stress Corrosion Cracking of Drip Shield and Waste Package Outer Barrier and Hydrogen Induced Corrosion of Drip Shield (CRWMS M&O 2000 [135773])	WP[d]	3.4-5 3.4-6 3.4-7 3.4-8
		Calculation of Probability and Size of Defect Flaws in Waste Package Closure Welds to Support WAPDEG Analysis (CRWMS M&O 2000 [144551])	WP[d]	3.4-9
Slow Rate of Radionuclide Mobilization and Release from the EBS	Radionuclide Inventory and Distribution in Repository	*Inventory Abstraction* (CRWMS M&O 2000 [136383])	WF[e]	3.5-8 3.5-9
	In-Package Environments	*In-Package Chemistry Abstraction* (CRWMS M&O 2000 [129287])	WF[e]	3.5-11
	Cladding Degradation and Performance	*Clad Degradation - Summary and Abstraction* (CRWMS M&O 2000 [147210])	WF[e]	3.5-18 3.5-19 4.1-9

Table 6.2-1. Summary of Analysis Model Reports, Process Model Reports, and Figures Illustrating Key Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Factor	Analysis Model Report	Process Model Report	Figure Illustrating Key Input Parameters or Intermediate Performance Results
Slow Rate of Radionuclide Mobilization and Release from the EBS (Continued)	CSNF Degradation and Performance	*CSNF Waste Form Degradation: Summary Abstraction* (CRWMS M&O 2000 [136060])	WF[e]	3.5-12
	DSNF Degradation and Performance	*DSNF and Other Waste Form Degradation Abstraction* (CRWMS M&O 2000 [144164])	WF[e]	3.5-13
	Defense HLW Degradation and Performance	*Defense High Level Waste Glass Degradation* (CRWMS M&O 2000 [143420])	WF[e]	3.5-14 3.5-15
	Dissolved Radionuclide Concentrations	*Summary of Dissolved Concentration Limits* (CRWMS M&O 2000 [143569])	WF[e]	3.5-21
	Colloid-Associated Radionuclide Concentrations	*Waste Form Colloid-Associated Concentrations Limits: Abstraction and Summary* (CRWMS M&O 2000 [125156])	WF[e]	3.5-24
	In-Package Radionuclide Transport	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	EBS[c]	
	EBS (Invert) Degradation and Performance	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	EBS[c]	4.1-10 4.1-12 4.1-14 4.1-15
		Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux (CRWMS M&O 2000 [152204])	NFE[b]	
Long Transport away from the EBS	UZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion)	*Unsaturated and Saturated Transport Properties* (CRWMS M&O 2000 [141440])	UZ[a]	
		Abstraction of Flow Fields for RIP (ID: U0125) (CRWMS M&O 2000 [123913])	UZ[a]	
		Particle Tracking Model and Abstraction of Transport Processes (CRWMS M&O 2000 [141418])	UZ[a]	3.7-9 3.7-10 3.7-11 4.1-18 4.1-19
		UZ Colloid Transport Model (CRWMS M&O 2000 [122799])	UZ[a]	

Table 6.2-1. Summary of Analysis Model Reports, Process Model Reports, and Figures Illustrating Key Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Factor	Analysis Model Report	Process Model Report	Figure Illustrating Key Input Parameters or Intermediate Performance Results
Long Transport away from the EBS (Continued)	Coupled Effects on UZ Radionuclide Transport	*Unsaturated Zone and Saturated Zone Transport Properties* (CRWMS M&O 2000 [141440])	UZ[a]	
	SZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion)	*Uncertainty Distribution for Stochastic Parameters* (CRWMS M&O 2000 [147972])	SZ[f]	
		Input and Results of the Base Case Saturated Flow and Transport Model for TSPA (CRWMS M&O 2000 [139440])	SZ[f]	3.8-18 3.8-19 4.1-20
	Wellhead Dilution	*Groundwater Usage by the Proposed Farming Community* (CRWMS M&O 2000 [144056])	BIO[g]	3.9-9
	Biosphere Dose Conversion Factors	*Distribution Fitting to the Stochastic BDCF Data* (CRWMS M&O 2000 [144055])	BIO[g]	
		Abstraction of BDCF Distributions for Irrigation Period (CRWMS M&O 2000 [144054])	BIO[g]	3.9-11
Minimal Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	*Characterize Framework for Igneous Activity at Yucca Mountain, Nevada (T0015)* (CRWMS M&O 2000 [141044])	DE[h]	
	Characteristics of Volcanic Eruption	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	DE[h]	
	Effects of Volcanic Eruption	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	DE[h]	
	Atmospheric Transport of Volcanic Eruption	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	DE[h]	
	Biosphere Dose Conversion for Volcanic Eruption	*Disruptive Event Biosphere Dose Conversion Factor Analysis* (CRWMS M&O 2000 [143378])	BIO[g]	
	Probability of Igneous Intrusion	*Characterize Framework for Igneous Activity at Yucca Mountain, Nevada (T0015)* (CRWMS M&O 2000 [141044])	DE[h]	
	Characteristics of Igneous Intrusion	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	DE[h]	

Table 6.2-1. Summary of Analysis Model Reports, Process Model Reports, and Figures Illustrating Key Input Parameters to Total System Performance Assessment-Site Recommendation (Continued)

Key Attributes of System	Factor	Analysis Model Report	Process Model Report	Figure Illustrating Key Input Parameters or Intermediate Performance Results
	Effects of Igneous Intrusion	*Igneous Consequence Modeling for TSPA-SR* (CRWMS M&O 2000 [139563])	DE[h]	

NOTES: [a] UZ = *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774])

[b] NFE = *Near-Field Environment Process Model Report* (CRWMS M&O 2000 [146589])

[c] EBS = *Engineered Barrier System Degradation, Flow, and Transport Process Model Report* (CRWMS M&O 2000 [145796])

[d] WP = *Waste Package Degradation Process Model Report* (CRWMS M&O 2000 [138396])

[e] WF = *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332])

[f] SZ = *Saturated Zone Flow and Transport Process Model Report* (CRWMS M&O 2000 [145738])

[g] BIO = *Biosphere Process Model Report* (CRWMS M&O 2000 [151615])

[h] DE = *Disruptive Events Process Model Report* (CRWMS M&O 2000 [141733])

N/A = not applicable

In conclusion, the data, analyses, and models used as the technical basis for the TSPA-SR, as well as the assumptions, uncertainty, variability, and conservatism that go along with these data, analyses, and models are all traceable back to their source documents and data sets. This traceability allows all interested reviewers to examine the defensibility of the individual component models and reach their own conclusions regarding their scientific adequacy.

6.2.2 Summary of Uncertainty Treatment in TSPA-SR Analyses

Total system performance assessments are by their very nature uncertain projections of the possible behavior of the individual component models describing the relevant processes affecting the containment and isolation of radioactive wastes from the biosphere. This uncertainty is explicitly included in the models and resulting analyses in the form of discrete probability distributions that encompass the range of possible outcomes.

As noted throughout this document and within the individual analysis model reports and process model reports that provide the technical basis for the TSPA-SR model, there remains uncertainty in the individual process models and their abstraction into the TSPA-SR model. Much of this uncertainty has been quantified and is included in the TSPA-SR model. The TSPA-SR results reflect this quantified uncertainty. For example, the full distribution of individual dose rates illustrated on Figure 6.1-1 indicate a range of possible dose rates that extends over about 6 orders of magnitude between the 5th and 95th percentile. This range is a direct indication of the degree of uncertainty incorporated in the individual models used as input to the TSPA-SR model.

In addition to the quantified uncertainty in the TSPA-SR model, there is also unquantified uncertainty that has been generally represented by using a more bounded or conservative representation of a particular process model. These conservatisms result when there is insufficient information available or significant complexity exists that is not amenable to quantified uncertainty (although elicitation approaches could be used if one desired to quantify the uncertainty in these conservative judgments). These conservatisms are summarized in Appendix F.

Simply acknowledging the nature and magnitude of uncertainty is one aspect of a performance assessment. However, more important is the assessment of the significance of that uncertainty in the ability of the site and engineered barriers to meet performance objectives. Chapter 5 of this document focused on the significance of the quantified uncertainty on the total system performance. (Note: The principal means for investigating the significance of unquantified uncertainties is through the use of alternative representations in "what-if" analyses. The content of this document has focussed on the quantified uncertainties. Unquantified uncertainties have been investigated as part of the barrier importance analyses conducted in support of the Repository Safety Strategy Rev 04, which are documented in *Repository Safety Strategy Revision 4* [CRWMS M&O 2000 [151001]]). In addition, continuing efforts are ongoing to evaluate the significance of conservatisms and other unquantified uncertainties. The objective of these analyses is to quantify the degree of conservatism in the total system performance results represented by the "base-case" models.

The quantitative uncertainty analyses have statistical evaluations of significance (such as regression analyses and classification analyses) documented in Section 5.1 and sensitivity and barrier importance analyses documented in Section 5.2 and 5.3, respectively. The distinction between sensitivity and barrier importance analyses is slight. In the former, only one model component (and in most cases only one parameter within one model component) is fixed at either the optimistic or pessimistic end of the uncertainty distribution to see the effect of that parameter or model on the total system performance. In barrier importance analyses, several model components (or parameters of several model components) are fixed at extreme values of their uncertainty distributions to see their effect on the system response.

Table 6.2-2 summarizes the sensitivity and barrier importance analyses conducted for the TSPA-SR. (Note: Additional barrier importance analyses conducted in support of the Repository Safety Strategy Rev 04 are documented in *Repository Safety Strategy Revision 4* [CRWMS M&O 2000 [151001]]). This table depicts the figure number in Section 5.2 or 5.3 which illustrates the significance of the quantified uncertainty on the overall system response. These figures generally illustrate the robustness of the overall repository response even when some components or barriers are fixed at their most pessimistic values within the uncertainty distribution that is believed most defensible in the corresponding analysis model report.

Table 6.2-2. Summary of Sensitivity and Barrier Importance Analyses in TSPA-SR

Key Attributes of System	Process Model Factor	Figure Illustrating Sensitivity Analysis	Figure Illustrating Barrier Importance Analysis
Water Contacting Waste Package	Climate		
	Net Infiltration	5.2-1	
	UZ Flow		5.3-1 5.3-2
	Coupled Effects on UZ Flow		
	Seepage into Emplacement Drifts	5.2-2	
	Coupled Effects on Seepage		
Waste Package Lifetime	In-Drift Physical and Chemical Environments		5.3-3
	In-Drift Moisture Distribution		
	Drip Shield Degradation and Performance	5.2-12 5.2-13	5.3-4 5.3-5
Waste Package Lifetime (Continued)	Waste Package Degradation and Performance	5.2-3 5.2-4 5.2-5 5.2-6 5.2-7 5.2-8 5.2-9 5.2-10 5.2-11	5.3-6 5.3-7
Radionuclide Mobilization and Release from the EBS	Radionuclide Inventory	5.2-14	
	In-Package Environments		
	Cladding Degradation and Performance		5.3-8
	Commercial Spent Nuclear Fuel Degradation and Performance		
	DOE-owned spent nuclear fuelDSNF Degradation and Performance		
	Defense high level radioactive waste Degradation and Performance		
	Dissolved Radionuclide Concentrations		5.3-9 5.3-10 5.3-11
	Colloid-Associated Radionuclide Concentrations		
	In-Package Radionuclide Transport		
	EBS (Invert) Degradation and Performance	5.2-22	

Table 6.2-2. Summary of Sensitivity and Barrier Importance Analyses in TSPA-SR (Continued)

Key Attributes of System	Process Model Factor	Figure Illustrating Sensitivity Analysis	Figure Illustrating Barrier Importance Analysis
Transport Away from the EBS	UZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion; Dilution)		5.3-12 5.3-13 5.3-14
	SZ Radionuclide Transport		5.3-15
	Wellhead Dilution		
	Biosphere Dose Conversion Factors	5.2-23 5.2-24	N/A
Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	5.2-25	N/A
	Characteristics of Volcanic Eruption	5.2-29 5.2-30	
	Effects of Volcanic Eruption	5.2-30	
	Atmospheric Transport of Volcanic Eruption	5.2-26 5.2-27	
	Biosphere Dose Conversion for Volcanic Eruption	5.2-28 5.2-33	
	Probability of Igneous Intrusion	5.2-25	
	Characteristics of Igneous Intrusion	5.2-31 5.2-32	
	Effects of Igneous Intrusion		

The figures cited on Table 6.2-2, along with the discussions in Chapter 5, also indicate the fact that when a particular uncertain parameter is fixed at an extreme value (i.e., either the 5th or 95th percentile of the distribution) the resulting variance of the projected dose is reduced. This variance reduction is a function of the degree the underlying parameter contributes to the overall variance in the base case analysis. For example, a significance fraction of the total variance of the dose at 100,000 years is the result of the wide distribution of waste package failures. Fixing the failures over a much narrower distribution reduces the variance of the projected dose.

Inherent uncertainties exist in any projection of the future performance of a deep geologic repository. These uncertainties must be considered in a demonstration of compliance with radiation protection standards that require waste containment for thousands of years. Many, but not all, of those uncertainties have been quantified and addressed in the TSPA; examples include:

- Potential changes in climate, seismicity, and other processes over the long compliance period for geologic disposal (i.e., 10,000 years)
- Variability and lack of knowledge of the properties of geologic media (e.g., heterogeneous permeability and porosity) over large spatial scales of the hydrogeologic setting (e.g., 20-km [12.5-mi] flow path from the repository to the point of compliance)

- Incomplete knowledge about the long-term material behavior of engineered components (e.g., corrosion of metals over many thousands of years).

Both the EPA and the NRC have recognized that uncertainty about the future performance of the repository will remain even after site characterization is complete. The NRC perspective is articulated in proposed 10 CFR Part 63, which states, "Proof that the geologic repository will be in conformance with the objective for postclosure performance is not to be had in the ordinary sense of the word" (64 FR 8640 [101680], Section IX, p. 8650). In place of such proof, the NRC regulation makes the determination of compliance with the standards contingent upon a regulatory finding of "reasonable assurance," which would be made on the basis of the record before it. Similarly, the EPA explicitly states that "unequivocal numerical proof of compliance is neither necessary nor likely to be obtainable" (64 FR 46976 [105065], p. 46997). The EPA prescribes a "reasonable expectation" approach for demonstrating compliance with its standard.

As the National Research Council (1990 [100061], p. 13) and others have noted, there are residual uncertainties with deep geologic disposal that cannot easily be quantified and incorporated into performance analyses. Nevertheless, their potential impact must be, to the extent practicable, addressed and, if important, mitigated to provide confidence in post-closure safety. Examples of residual uncertainties associated with geologic disposal that are difficult to quantify include:

- The potential for currently unknown processes to affect performance

- The possibility that incompletely characterized processes have been incorporated in the TSPA in a manner that results in the underestimation of radionuclide releases; examples of incompletely characterized processes include thermal, chemical, hydrologic, and mechanical processes that are coupled in complex ways that cannot be completely tested at the scale of a repository, and processes that are difficult to observe and test, such as colloidal transport of radionuclides

- Uncertainty associated with the projections of engineered barrier performance over geologic time periods (e.g., 10,000-years) based on data from short-term (e.g., several years) laboratory testing

- Uncertainty associated with the large spatial scale of the three-dimensional groundwater flow system, which makes it difficult to characterize flow paths and processes.

Substantial effort has been made to identify, characterize, and mitigate the potential impacts of residual uncertainties that could significantly affect long-term performance. Where possible, more tests have been conducted to collect additional information that would provide insight to analysts. Where additional testing was not feasible (e.g., it is not possible to run tests over the same time period as the repository must perform), or of limited benefit (e.g., no amount of excavation or drilling could completely characterize the natural system) modelers used conservative assumptions to "bound" their analyses of uncertain processes. To do this, they have incorporated assumptions in their models that represent the range of properties and processes that they believe are feasible. In addition, use has been made of empirical observations and qualitative lines of evidence from natural analogues to address uncertainties.

In conclusion, addressing the uncertainty inherent in the models used for post-closure performance assessment has been an integral ingredient in the development of the component models and parameters used in the TSPA-SR model. In addition, examining the significance of quantified uncertainty has been an important objective of the TSPA analyses themselves. Ongoing efforts to quantitatively evaluate the significance of unquantified uncertainties will add additional insights to support the determination of the degree of conservatisms included in the "base case" models described in Section 3.

6.2.3 Summary of Technical Issues Addressed in TSPA-SR Model and Analyses

The TSPA-SR is the fifth major iteration of TSPAs conducted by the DOE over the past decade in support of evaluating the suitability of the Yucca Mountain site and engineered barriers. The first three iterations of the TSPAs (Barnard, et al. 1992 [100309]; CRWMS M&O 1994 [100111]; CRWMS M&O 1995 [100198]) focussed on developing, implementing and testing the approach and methodology for performing a TSPA and on identifying the key information needs of those component models that were most significant contributors to system performance. The fourth iteration (DOE 1998 [100550], Volume 3) was conducted in support of the DOE Viability Assessment. Each of these previous iterations has benefited from insights, reviews, and criticisms developed from the predecessor analyses. As the scientific information available to evaluate the performance of the potential Yucca Mountain repository system has evolved over time, so too have the TSPA analyses.

The current TSPA-SR Rev 00 has benefited from reviews of the TSPA-VA completed by a Peer Review Panel (see Budnitz et al. 1999 [102726]), the NRC (Paperiello 1999 [146561]), Clark County, NV (Cohen 1999 [151783]), and the U.S. Geological Survey (Anderson et al. 1998 [101656]). It is anticipated that the TSPA-SR Rev 01 (to be completed following the completion of the Site Recommendation Consideration Report) will benefit from reviews conducted by similar review groups, including the Nuclear Waste Technical Review Board, as well as the public.

While it is difficult to enumerate every comment on the TSPA-VA, Appendix H presents a summary of many of the most significant comments and how they have been addressed. However, this comment resolution correlation matrix does not completely represent the breadth of comments received on all aspects of the VA. For example, numerous issues associated with the models included in the TSPA-VA have been identified in the NRC's Key Technical Issues Issue Resolution Status Reports (IRSRs) as Acceptance Criteria for evaluating the sufficiency of the DOE site recommendation. These issues are addressed in the individual PMRs that most closely correlate to the corresponding key technical issues.

While the models and analyses that support the TSPA-SR may not have addressed every issue or comment raised on the previous TSPA iterations, they have addressed the most significant issues. For example, a significant cross-cutting issue raised on the TSPA-VA was the reliance on expert elicitation in the absence of sufficient site or engineering data. In the present analysis, with the exception of volcanic and seismic hazards which are most amenable to elicitation methods and a use of TSPA-VA elicitations to confirm the uncertainty in some parameters used in the SZ flow and transport model, there have been no elicitations used in lieu of site data.

Another criticism on the TSPA-VA was related to the lack of traceability of the TSPA to the underlying data and analyses. In the present TSPA-SR, this traceability has been significantly enhanced by the tracking of specific AMRs and data tracking numbers generated under common process controls by all scientific investigators. An example of this traceability is the information flow presented in Appendix E. Another example is the listing of model runs included as Appendix G.

A cross-cutting comment on the TSPA-VA was the treatment of uncertainty in the models and analyses used to support the TSPA. In particular it was noted by the TSPA-VA Peer Review Panel (Budnitz et al. 1999 [102726]) that an analysis of the future "probable" behavior of the proposed repository may be beyond the analytical capability. They go on to acknowledge that various approaches can be used to evaluate the complex and difficult to analyze processes that comprise the total system model. These approaches include (Budnitz et al., 1999 [102726]): (1) updating the component models, (2) expanding the quality and quantity of data available as input to these analyses, (3) using bounding analyses (i.e., intentionally conservative assumptions, parameters and models), and (4) design changes.

All of these approaches have been embodied in the TSPA-SR. All models, analyses and data used to support the TSPA-SR have been substantially improved over those used in the development of the TSPA-VA. In addition, the same quality processes have been used in the development, testing, checking and review of the products (in particular the Analysis/Model Reports) used to support the TSPA-SR model. Also, intentionally conservative assumptions have been utilized in those areas of large conceptual uncertainty. Examples of areas with intentionally conservative assumptions include near-field coupled process effects, and in-drift and in-package thermal hydrology and radionuclide transport. Finally, the design analyzed in the TSPA-SR has several attributes that are included to mitigate the effects of some uncertainties; for example a drip shield that minimizes the effects of seepage, a wider drift spacing and closer waste package spacing to minimize localized effects of heat-induced water mobilization and a cooler operating mode to minimize some of the near-drift coupled process effects.

In conclusion, the comments and criticisms made on previous TSPA iterations have improved the methodology and approach and their implementation in this current analysis. The models used to support the TSPA have been significantly enhanced to reflect the most current understanding of the features, events and processes relevant to the post-closure performance. The family of models and analyses have been developed and controlled through a formal process to assure they are adequate for their use in projecting the long-term performance. Finally, the documentation of the models, their integration and the resulting post closure projections of performance have been enhanced to more clearly illustrate the behavior of the system.

6.3 SUMMARY OF HOW AND WHERE TSPA-SR HAS ADDRESSED THE PROPOSED REGULATORY OBJECTIVES

The requirements for a performance assessment (as defined in proposed 10 CFR Part 63 [64 FR 8640 [101680]]) or a total system performance assessment (as defined in proposed 10 CFR Part 963 [64 FR 67054 [124754]]) have been specified in the applicable proposed regulations and in Acceptance Criteria in the Total System Performance Assessment and Integration key technical issues in the Issue Resolution Status Report. This section reiterates these requirements and enumerates where they have been addressed in this TSPA-SR report or the family of AMRs and PMRs that provide the technical foundation for the TSPA-SR. (Note: This section does not present the compliance argument with the requirements and criteria specified in proposed 10 CFR 963.16 and 963.17 [64 FR 67054 [124754]]). The compliance demonstration for these two sections of Part 963 is being presented in the *Site Recommendation Consideration Report,* which is currently being developed.

6.3.1 Regulatory Objectives of Proposed 10 CFR Part 63 (64 FR 8640 [101680])

The requirements for performance assessment are specified in proposed 10 CFR 63.114 (64 FR 8640 [101680]), as follows:

> Any performance assessment used to demonstrate compliance with 63.113(b) shall:
>
> (a) include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site, and the surrounding region to the extent necessary, and information on the design of the EBS, used to define parameters and conceptual models used in the assessment,

This requirement has been addressed in the analyses and models included in the TSPA-SR that are summarized in (1) Section 3; (2) the nine PMRs and (3) the over one hundred AMRs that provide the "family tree" for the TSPA-SR as indicated in Appendix E. This body of information is extensive and the interested reader is referred to the appropriate PMR or AMR that has been cited throughout this technical document.

> (b) Accounts for uncertainties and variabilities in parameter values and provide the technical basis for parameter ranges, probability distributions or bounding values used in the performance assessment,

This requirement has been addressed in the analyses and models summarized in Chapter 3 and the nine process model reports. Full parameter distributions are summarized in Chapter 3 and are presented in greater detail in the TSPA-SR Model document (CRWMS M&O 2000 [148384]). In some cases the uncertainty or variability has not been explicitly quantified but has been addressed by more bounding approximations. This is particularly true for complex coupled processes for which there is limited site-specific information over the spatial and temporal time scales of interest.

> (c) Consider alternative conceptual models of features and processes that are consistent with available data and current scientific understanding, and evaluate the effects that alternative conceptual models have on the performance of the geologic repository,

This requirement has been addressed in the over one hundred analysis model reports that support the TSPA-SR "family tree" as illustrated in Appendix E. In these documents, the effects of alternative conceptual models are addressed primarily by their effect on some component process model or abstraction rather than their effect on the performance of the repository system as a whole. Additional analyses to quantify the effect of unquantified alternative conceptual models may still be required to fully address this requirement.

(d) Consider only events that have at least one chance in 10,000 of occurring over 10,000 years,

This requirement has been addressed through the features, events and process screening process, the results of which are summarized in Appendix B and the nine process model reports.

(e) Provide the technical basis for either inclusion or exclusion of specific features, events, and processes of the geologic setting in the performance assessment. Specific features, events, and processes of the geologic setting must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

This requirement has been addressed through the features, events and process screening process, the results of which are summarized in Appendix B and the nine process model reports.

(f) Provide the technical basis for either inclusion or exclusion of degradation, deterioration or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

This requirement has been addressed in Sections 3.3 through 3.6 and the cited analysis model reports and process model reports that relate to engineered barrier degradation processes.

(g) Provide the technical basis for models used in the performance assessment such as comparisons made with outputs of detailed process-level models and/or empirical observations (e.g., laboratory testing, field investigations, and natural analogs).

This requirement has been addressed in Chapter 3 and the TSPA-SR Model document (CRWMS M&O 2000 [148384]) as well as the individual abstraction models referenced in Chapter 3.

(h) Identify those design features of the EBS, and the natural features of the geologic setting, that are considered barriers important to waste

(i) Describe the capability of the barriers identified as important to waste isolation to isolate waste, taking into account uncertainties in characterizing and modeling the barriers.

This requirement has been addressed by the identification of barriers included in Section 5.3. In addition, Table 6.3-1 below illustrates the correlations between the individual barriers and the process model factors presented in Section 3 and identifies the corresponding key attributes of the repository safety strategy. As indicated in Table 6.3-1, not all component models in the TSPA-SR model are barriers in the strict definition of the term (e.g., limiting water flow or radionuclide migration); many are simply required in order to have a complete total system model.

Table 6.3-1. Correlation of Barrier and Barrier Functions to Key Attributes of Yucca Mountain Repository System and Process Model Factors

<table>
<tr><th>Key Attributes of System</th><th>Process Model Factor</th><th>Barrier</th><th>Barrier Function</th></tr>
<tr><td rowspan="2">Limiting Water Contacting Waste Package</td><td>Climate</td><td rowspan="2">Surficial soils and topography</td><td rowspan="2">Reduce the amount of water entering the unsaturated zone by surficial processes (e.g., precipitation lost to runoff, evaporation, and plant uptake</td></tr>
<tr><td>Net Infiltration</td></tr>
<tr><td rowspan="4">Limiting Water Contacting Waste Package (Continued)</td><td>Unsaturated Zone Flow</td><td rowspan="4">Unsaturated rock layers overlying the repository and host unit</td><td rowspan="4">Reduce the amount of water reaching the repository by subsurface processes (e.g., lateral diversion and flow around emplacement drifts)</td></tr>
<tr><td>Coupled Effects on Unsaturated Zone Flow</td></tr>
<tr><td>Seepage into Emplacement Drifts</td></tr>
<tr><td>Coupled Effects on Seepage</td></tr>
<tr><td rowspan="4">Prolonging Waste Package Lifetime</td><td>In-Drift Physical and Chemical Environments</td><td rowspan="2">N/A</td><td rowspan="2">These factors provide conditions that affect performance, but are not barriers per se in the TSPA-SR.</td></tr>
<tr><td>In-Drift Moisture Distribution</td></tr>
<tr><td>Drip Shield Degradation and Performance</td><td>Drip shield around the waste packages</td><td>Prevent water contacting the waste package and waste form by diverting water flow around the waste package; therefore limiting advective transport through the invert</td></tr>
<tr><td>Waste Package Degradation and Performance</td><td>Waste package</td><td>Prevent water from contacting the waste form</td></tr>
<tr><td rowspan="10">Limiting Radionuclide Mobilization and Release from the Engineered Barrier System</td><td>Radionuclide Inventory</td><td rowspan="2">N/A</td><td rowspan="2">These factors provide conditions that affect performance, but are not barriers per se in the TSPA-SR.</td></tr>
<tr><td>In-Package Environments</td></tr>
<tr><td>Cladding Degradation and Performance</td><td>Spent fuel cladding</td><td>Delay and/or limit liquid water contacting spent nuclear fuel after waste packages have degraded</td></tr>
<tr><td>CSNF Degradation and Performance</td><td rowspan="6">Waste form</td><td rowspan="7">Limit radionuclide release rates as a result of low solubilities or low diffusion through degraded engineered barriers</td></tr>
<tr><td>DSNF Degradation and Performance</td></tr>
<tr><td>DHLW Degradation and Performance</td></tr>
<tr><td>Dissolved Radionuclide Concentrations</td></tr>
<tr><td>Colloid-Associated Radionuclide Concentrations</td></tr>
<tr><td>In-Package Radionuclide Transport</td></tr>
<tr><td>EBS (Invert) Degradation and Performance</td><td>Drift Invert</td></tr>
</table>

Table 6.3-1. Correlation of Barrier and Barrier Functions to Key Attributes of Yucca Mountain Repository System and Process Model Factors (Continued)

Key Attributes of System	Process Model Factor	Barrier	Barrier Function
Slow Transport Away from the Engineered Barrier System	Unsaturated Zone Radionuclide Transport (Advective Pathways; Retardation; Dispersion; Dilution)	Unsaturated rock layers below the repository	Delay radionuclide movement to the groundwater aquifer because of water residence time, matrix diffusion, and/or sorption
	Saturated Zone Radionuclide Transport	Tuff and alluvial aquifers (flow path extending from below the repository to point of compliance)	Delay radionuclide movement to the receptor location by water residence time, matrix diffusion, and/or sorption
	Wellhead Dilution	N/A	These factors provide conditions that affect performance but are not barriers per se.
	Biosphere Dose Conversion Factors		
Addressing Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	N/A	These factors provide conditions that affect performance but are not barriers.
	Characteristics of Volcanic Eruption		
	Effects of Volcanic Eruption		
	Atmospheric Transport of Volcanic Eruption		
	Biosphere Dose Conversion for Volcanic Eruption		
	Probability of Igneous Intrusion		
	Characteristics of Igneous Intrusion		
	Effects of Igneous Intrusion		

This requirement has been addressed in the barrier importance analyses presented in Section 5.3. Additional information on this topic may be found in the Repository Safety Strategy Rev 04 (CRWMS M&O 2000 [148713]).

(j) Provide the technical basis for the description of the capability of the barriers, identified as important to waste isolation, to isolate waste.

This requirement has been addressed in the barrier importance analyses presented in Section 5.3. Additional information on this topic may be found in the Repository Safety Strategy Rev 04 (CRWMS M&O 2000 [148713]).

In conclusion, the basic requirements for a performance assessment as specified in proposed 10 CFR Part 63 have been addressed with the suite of analyses and models that support the TSPA-SR, by the TSPA-SR model itself, and by the analyses described in the present technical document. This is not to imply that the potential Yucca Mountain repository is suitable for licensing. That is a decision that will only be made by the Secretary upon consideration of all information, including sufficiency reviews by NRC and comments from the public.

All of the above information and their integration in the context of this TSPA-SR provide a sound, traceable and transparent technical picture of the possible performance of a potential repository at Yucca Mountain. These projections have incorporated the best available science and technology developed over years of investigating the Yucca Mountain site and the associated waste forms and waste packages. Although significant uncertainty exists in some of the component models underlying the TSPA-SR (as identified in Section 3 and Appendix F), these uncertainties have either been reasonably quantified, or in some cases of great complexity, conservatively bounded. Therefore, it is reasonable to conclude that the expected performance, and the associated uncertainty in that performance, of a potential repository at Yucca Mountain has been captured in the suite of analyses presented in this document.

6.3.2 Regulatory Objectives in the Total System Performance Assessment and Integration Issue Resolution Status Report

The Issue Resolution Status Report on Total System Performance Assessment and Integration is the principal vehicle by which the NRC staff provides the DOE with feedback regarding Total System Performance Assessment and Integration issue resolution before the Site Recommendation and License Application. The Total System Performance Assessment and Integration Issue Resolution Status Report (NRC 2000 [151753], pp. 3 to 4) notes that:

> … a critical aspect of an acceptable TSPA is the integration of information from many technical disciplines in the modeling and abstraction of the engineered system and natural features, events and processes (FEPs). The need to adequately address this integration of technical disciplines in the development of a TSPA is specifically addressed in this [Issue Resolution Status Report]. The incorporation of acceptance criteria addressing the integration issues in this [Issue Resolution Status Report] is designed to ensure that in issue resolution and the eventual [License Application], the transfer of information among the technical disciplines and to DOE's TSPA occurs, the analysis is focused on the integrated total system assessment, and the assessment is transparent, traceable, defensible and comprehensive.

NOTE: A more recent revision (Revision 3) of the Total System Performance Assessment and Integration Issue Resolution Status Report (IRSR) has become available in September 2000. As of this writing, this document has not been evaluated in the context of the applicable acceptance criteria revised in this version of the IRSR. Future analyses will consider the changes made in this and any subsequent revisions of this document or the risk-informed performance-based Yucca Mountain Review Plan currently in development by NRC staff.

The Issue Resolution Status Report is divided into four subissues and several related acceptance criteria. The four subissues are:

- System description and demonstration of multiple barriers
- Scenario analysis
- Model abstraction
- Demonstration of the overall performance objective.

The following discussion briefly describes the status of how and where the acceptance criteria have been addressed in the TSPA-SR family of documents.

Two programmatic acceptance criteria, quality assurance and expert elicitation, are applicable to all the TSPA subissues. These acceptance criteria are:

> Criterion P1: The collection, documentation, and development of data, models, and/or computer codes have been performed under acceptable quality assurance procedures, or if the data, models, and/or computer codes were not subject to an acceptable [Quality Assurance] procedure, they have been appropriately qualified.

This acceptance criterion was addressed through the process of controlling all the information supporting the TSPA-SR model and all the analyses conducted with the TSPA-SR model that have been documented in this report. For example, all data have been controlled per QA procedure AP-SIII.3Q [149901], all models and analyses have been controlled per AP-3.10Q [151293], all calculations controlled per AP-3.11Q [153200], all technical reports controlled per AP-3.12Q [153122], and all software have been controlled per AP-SI.1Q [146376].

> Criterion P2: Formal expert elicitations can be used to support data synthesis and model development for DOE TSPA, provided that the elicitations are conducted and documented under acceptable procedures.

This criterion was addressed because the two expert elicitations that support the TSPA-SR (namely the Probabilistic Volcanic Hazard Assessment and the Probabilistic Seismic Hazard Assessment) were both performed under the applicable procedures that were driven by guidance provided in NUREG. In addition, input from an expert elicitation conducted to support the TSPA-VA were used to support the determination of the uncertainty in some parameters of the SZ flow and transport model.

The Acceptance Criteria related to the System Description and Demonstration of Multiple Barriers subissue include those that relate to:

1. Transparency and traceability of the analysis, including
    a. TSPA documentation style, structure, and organization
    b. Features, events and processes identification and screening
    c. Abstraction methodology
    d. Data use and validity
    e. Assessment results
    f. Code design and data flow.
2. Demonstration of multiple barriers.

The Acceptance Criteria related to the TSPA Methodology of Scenario Analysis subissue include those that relate to:

1. Identification of an initial set of processes and events
2. Classification of processes and events

3. Screening of processes and events
4. Formation of scenarios
5. Screening of scenario classes.

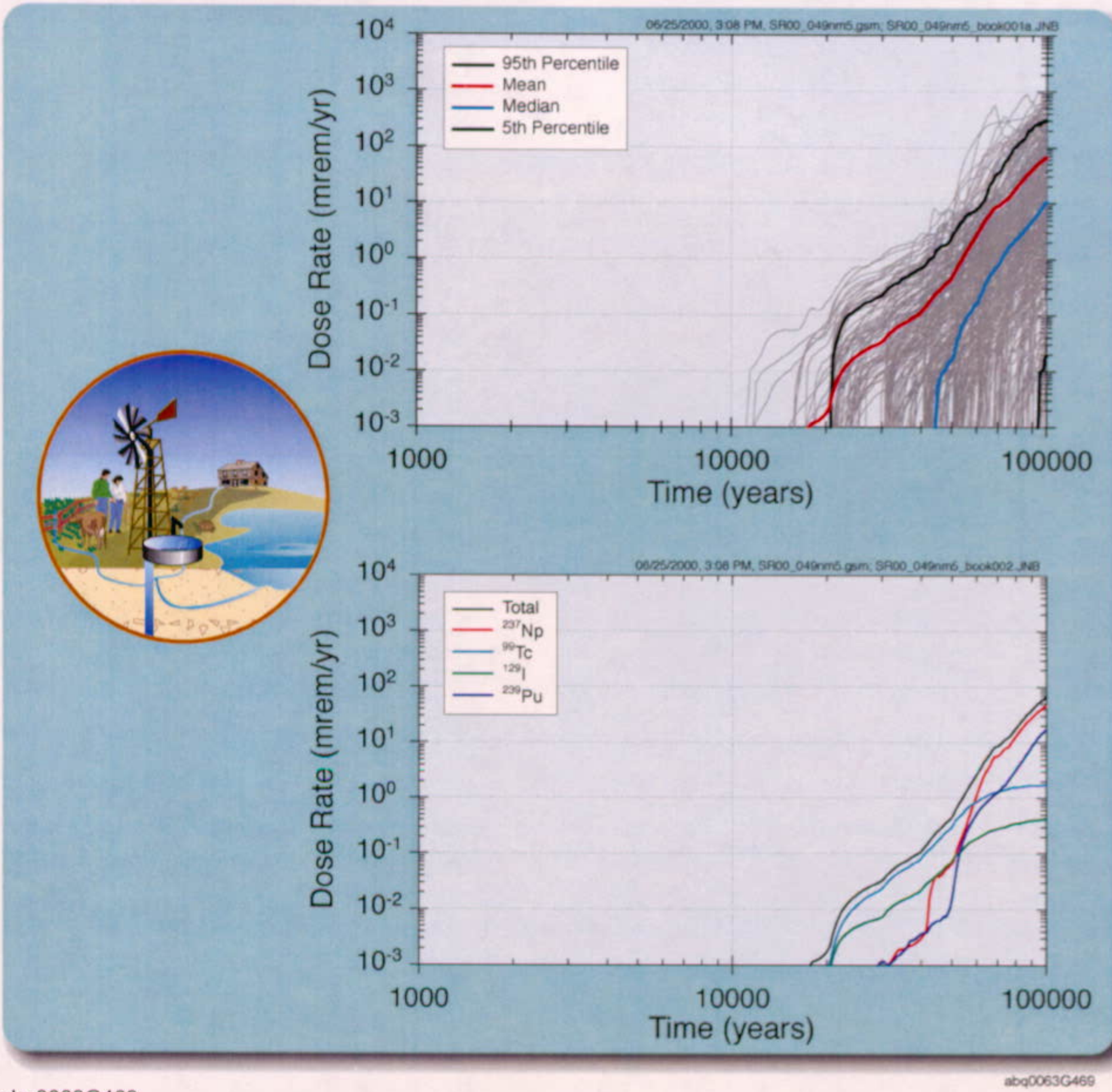
The Acceptance Criteria related to the TSPA Methodology of Model Abstraction subissue apply to all component models included in the TSPA model. These criteria include those that relate to:

1. Data and model justification
2. Data uncertainty
3. Model uncertainty
4. Model support
5. Integration.

The Acceptance Criteria related to the Demonstration of the Overall Performance Objective subissue are not yet available. They will be established by NRC after proposed 10 CFR Part 63 (64 FR 8640 [101680]) is published in final form.

Each of the above acceptance criteria are addressed in Appendix D. For the current state of FEPs screening and scenario development; model abstraction development and documentation; TSPA model integration; and the treatment and documentation of the uncertainty associated with the TSPA analysis, these acceptance criteria have been adequately addressed. Again, that is not to imply that no remaining analysis or model development is required prior to licensing, if the Yucca Mountain site is found suitable and is recommended to the President.

In conclusion, the documentation of the TSPA-SR, including the analysis model reports, process model reports and the TSPA-SR model document, provide the technical basis to address the NRC acceptance criteria in the Total System Performance Assessment and Integration Issue Resolution Status Report, as well as the TSPA-related acceptance criteria in the other key technical issues Issue Resolution Status Reports. In addition, these same documents provide the scientific basis for evaluating the suitability of the Yucca Mountain site. This suitability evaluation, which uses proposed 10 CFR Part 963 (64 FR 67054 [124754]) evaluation criteria, is being presented in the *Site Recommendation Consideration Report* currently under development.



abq0063G469

Figure 6.1-1. Summary of Individual Protection Performance Results–Nominal Scenario Class

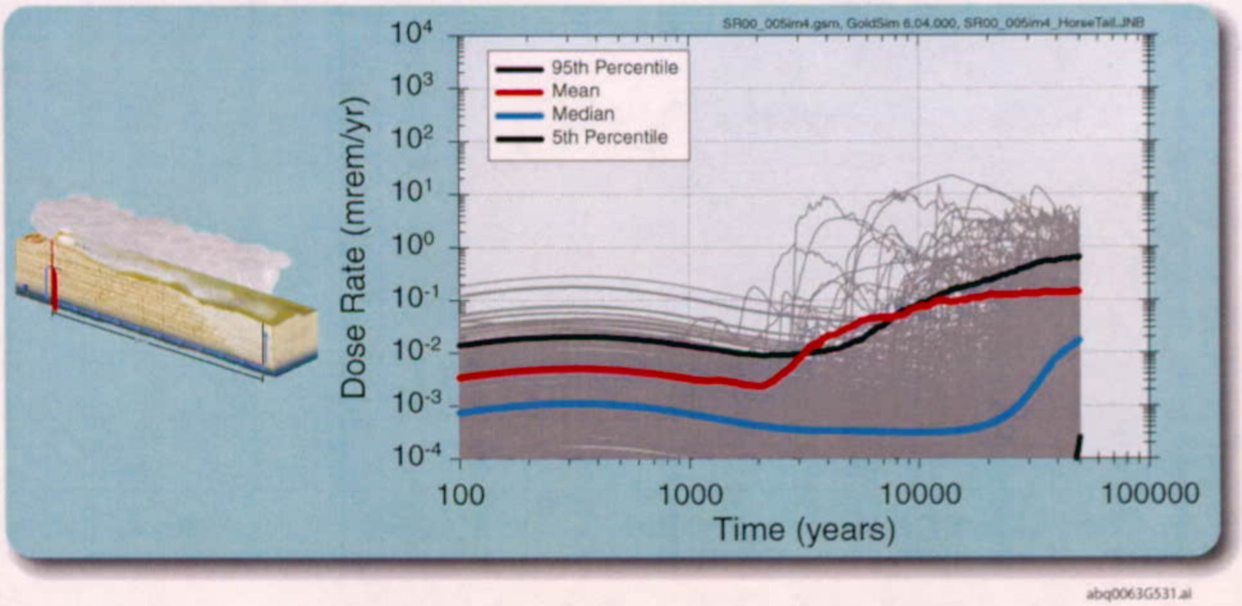


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Figure 6.1-2. Summary of Individual Protection Performance Results–Volcanic Scenario Class

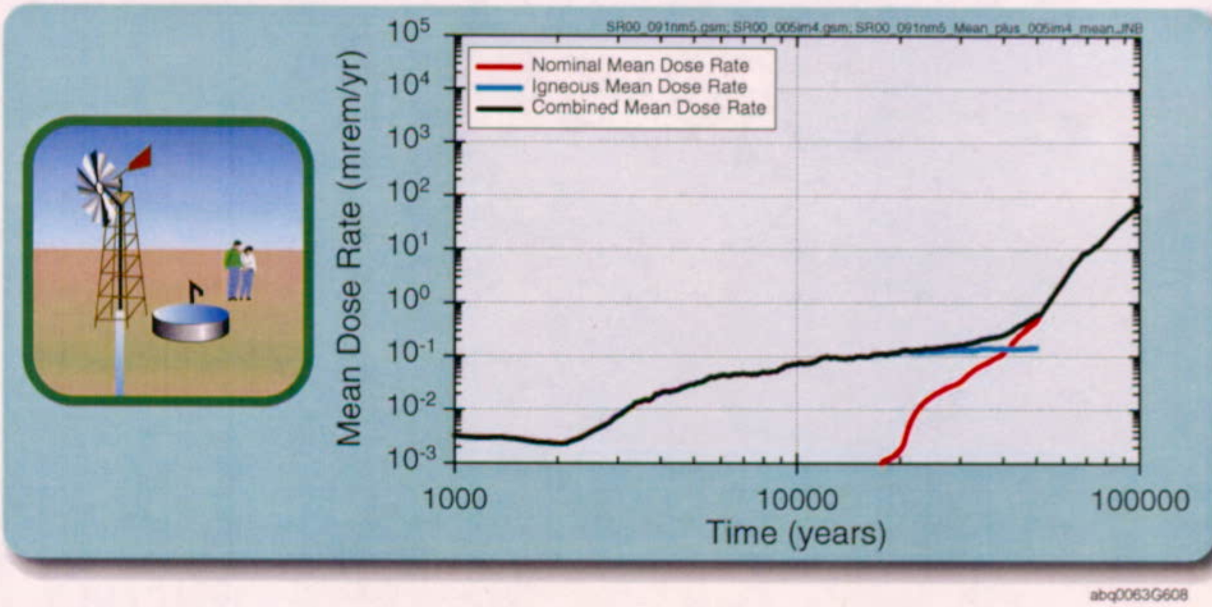

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Figure 6.1-3. Summary of Individual Protection Performance Results–Total System Combined Scenario Class

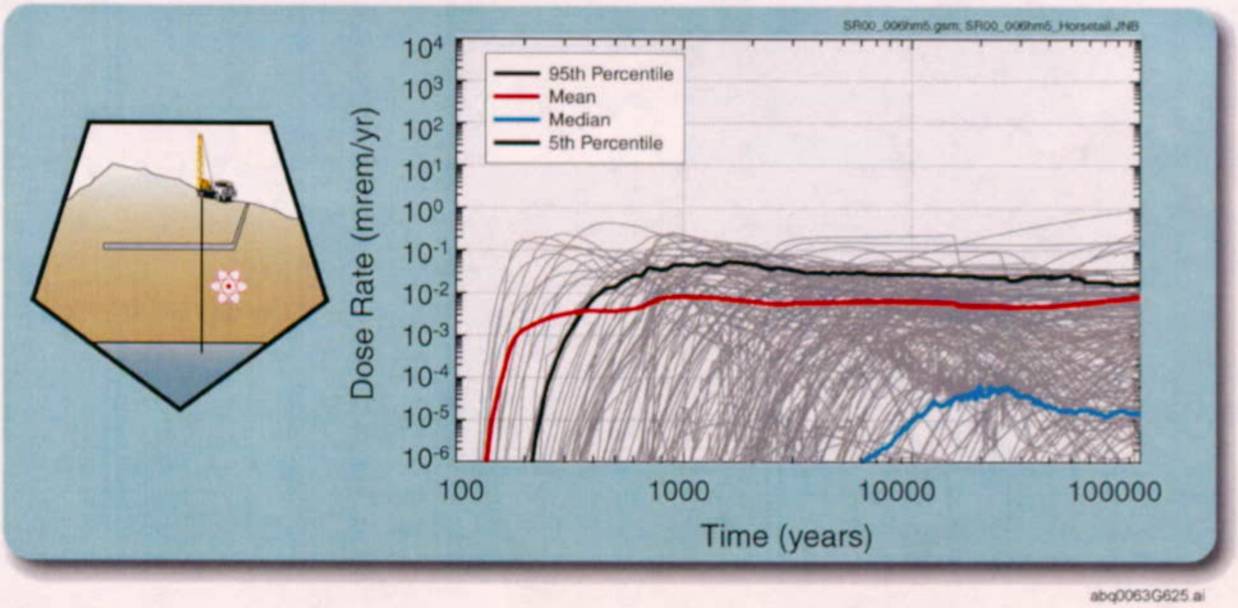

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Figure 6.1-4. Summary of Human Intrusion Performance Results–Assumed Human Intrusion Event Occurs at 100 Years

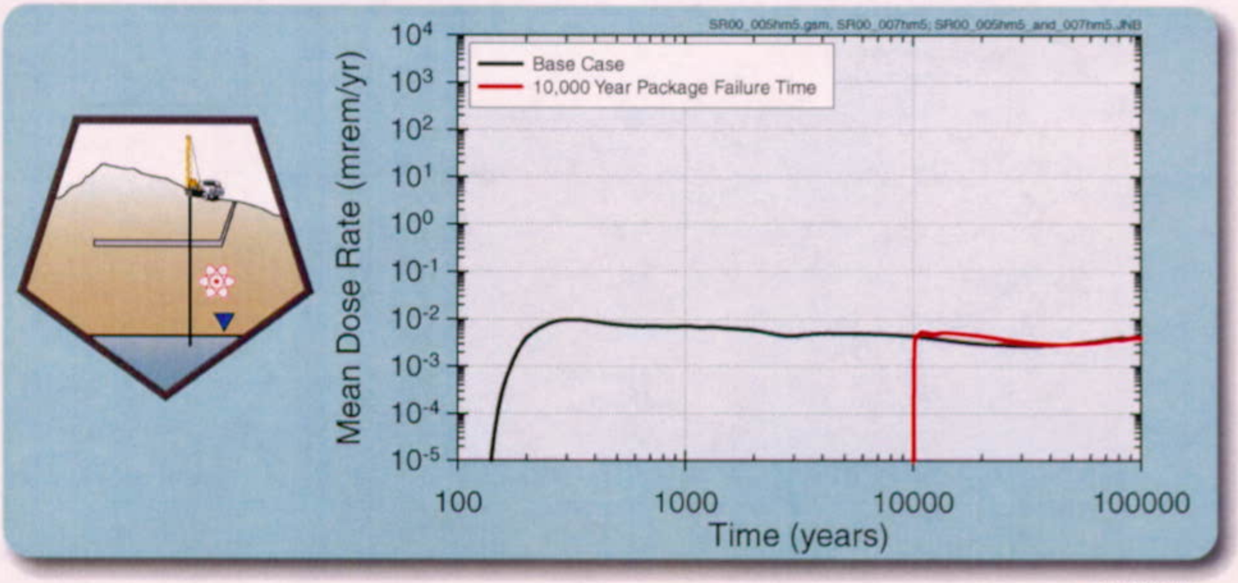

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Figure 6.1-5. Summary of Human Intrusion Performance Results–Assumed Human Intrusion Event Occurs at 10,000 Years

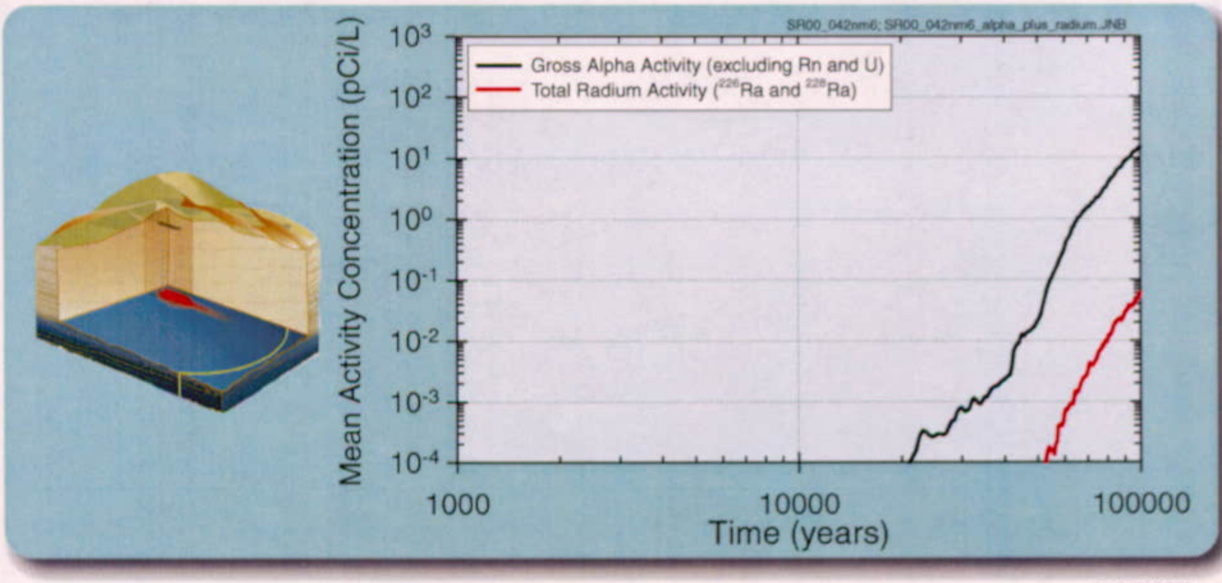

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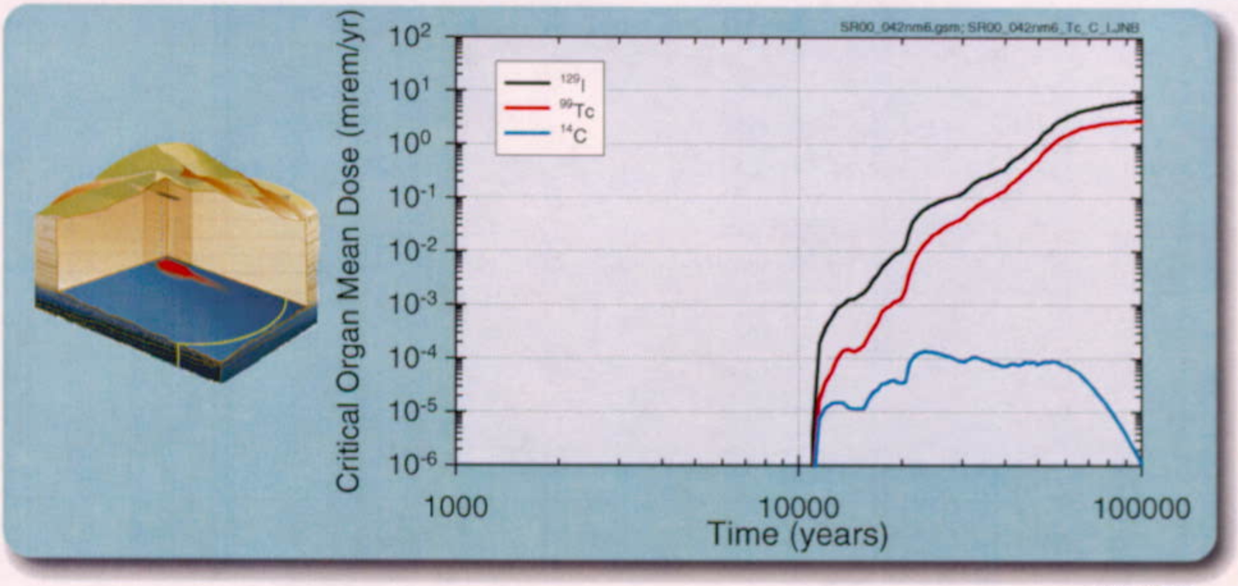
Figure 6.1-6. Summary of Groundwater Protection Performance Results–Gross Alpha Activity



abq0063G630

Figure 6.1-7. Summary of Groundwater Protection Performance Results–Combined Beta and Photon Emitting Radionuclides

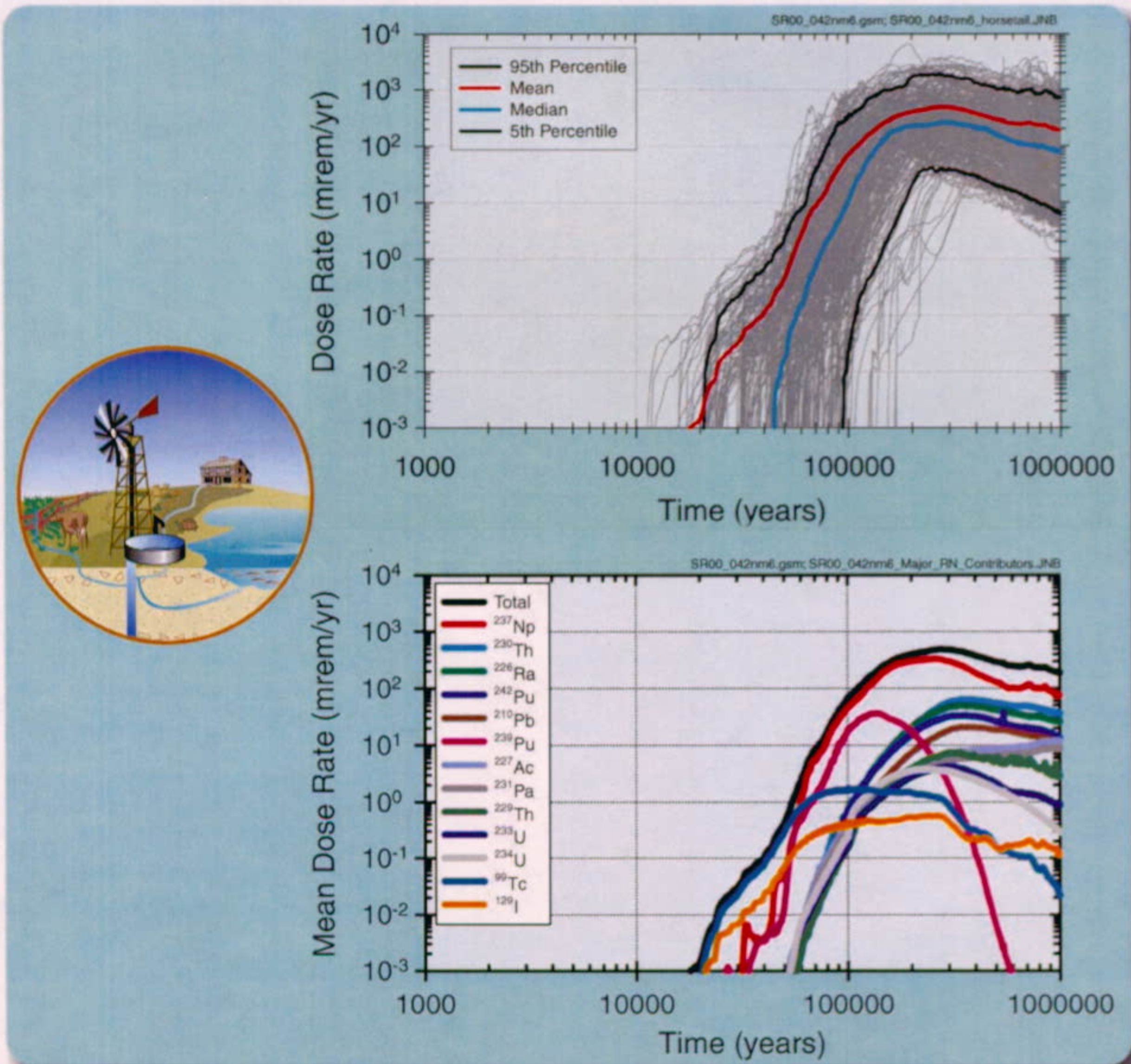


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Figure 6.1-8. Summary of Peak Dose Performance Results

7. REFERENCES

The following is a list of the references cited in this document. Column 1 represents the unique six digit DIRS number, which is placed in the text following the reference callout (e.g., CRWMS M&O 2000 [144054]). The purpose of these numbers is to assist the reader in locating a specific reference. Within the reference list, multiple sources by the same author (e.g., CRWMS M&O 2000) are ordered numerically by the DIRS number.

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7.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

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7.3 SOURCE DATA, LISTED BY TRACKING NUMBER

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149980 GS971000012847.004. Water Quality Data Collected from Springs and Wells in the Yucca Mountain Region from May 6, 1997 to May 15, 1997. Submittal date: 10/23/1997.

148751 LA0003AM831341.001. Preliminary Revision of Probability Distributions for Sorption Coefficients (K_DS). Submittal date: 03/29/2000.

149557 LA0003JC831362.001. Preliminary Matrix Diffusion Coefficients for Yucca Mountain Tuffs. Submittal date: 4/10/2000.

147285 LA0003MCG12213.002. Cumulative Probabilities for Colloid Transport Between One Matrix and Another Calculated from Interpolation of Pore Volume Data from Yucca Mountain Hydrologic (Stratigraphic) Samples. Submittal date: 03/10/2000.

149593 LA0004FP831811.002. Volume of Volcanic Centers in the Yucca Mountain Region. Submittal date: 04/14/2000.

151391 LA0004FP831811.004. Summary Frequencies of Disruptive Volcanic Events. Submittal date: 04/25/2000.

146932 LA9911GZ12213S.001. SZ Flow and Transport Model. Submittal date: 12/23/1999.

144279 LAFP831811AQ97.001. Chemical and Geochronology Data for the Revision and Final Publication of the Volcanism Synthesis Report. Submittal date: 08/29/1997.

141284 LL000112205924.112. Long Term Corrosion Test Facility Data. Submittal date: 01/25/2000.

142902 LL991109851021.095. Colloid Size and Concentration Investigations in Scientific Notebook SN 1381. Submittal date: 01/10/2000.

144927 LL991212305924.108. Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier. Submittal date: 12/20/1999.

144567 MO0001SPASUP03.001. Data to Support Calculation of Probability and Size of Defect Flaws in Waste Package Closure Welds to Support WAPDEG Analysis. CAL-EBS-PA-000003 REV 00. Submittal date: 01/31/2000.

150755 MO0002SPADVE03.001. Disruptive Volcanic Event BDCF. Submittal date: 02/14/2000.

149168 MO0002SPALOO46.010. Lookup Tables for PH, CL, and Ionic Strength Predicted by Precipitates/Salts Model for THC Abstraction. Submittal date: 02/07/2000.

148338 MO0002SPASDC00.002. Self-Diffusion Coefficient of Water. Submittal date: 02/24/2000.

150886 MO0003RIB00083.000. Dissolution Rate and Waste Form Degredation. Submittal date: 03/14/2000.

148872 MO0003SPAABS07.006. Abstracted BDCF Distributions with Soil Erosion for Use in TSPA-SR. Submittal date: 03/23/2000. Submit to RPC URN-0560

148453 MO0003SPAABS08.004. Abstracted BDCF Distributions for Use in TSPA-SR. Submittal date: 03/21/2000. Submit to RPC URN-0561

147949 MO0003SPAHIG12.002. Highest and Lowest Observed or Expected Masses of Iron-(hydr)Oxide Colloids Per Unit Volume or Mass of Water. Submittal date: 03/02/2000.

147952 MO0003SPAHLO12.004. Highest and Lowest Observed or Expected Groundwater Colloid Masses Per Unit Volume or Mass of Water; Values of Ionic Strength Above Which Groundwater Colloid Dispersions Are Unstable and Below Which Groundwater Colloid Dispersions Are Stable (Within Defined pH Range). Submittal date: 03/16/2000.

147951 MO0003SPAION02.003. Values Of Ionic Strength That Define The Stability Limits Of Iron-(Hydr)Oxide Colloids. Submittal date: 03/03/2000.

147953 MO0003SPALOW12.001. Lowest Observed or Expected Concentration of Radionuclide Element Rn Associated with Waste-Form Colloids. Submittal date: 03/02/2000.

148992 MO0003SPAPCC03.004. Supporting Media for Abstraction of Models for Pitting and Crevice Corrosion of Drip Shield and Waste Package Outer Barrier. Submittal date: 03/31/2000.

151075 MO0003SPASGU01.003. Stochastic Groundwater Usage in Amargosa Valley for TSPA-SR. Submittal date: 03/21/2000. Submit to RPC URN-0562

147299 MO0003SPASUP02.003. Supporting Media for Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis. Submittal date: 03/02/2000.

149092 MO0004MWDRIFM3.002. Results of the Yucca Mountain Probabilistic Seismic Hazard Analysis (PSHA). Submittal date: 04/14/2000.

148923 MO0004SPABDCFS.001. Preliminary Biosphere Dose Conversion Factors (BDCFS) to be Used in the TSPA for SR. Submittal date: 04/10/2000.

151368 MO0004SPACLD07.043. Clad Degradation - Summary and Abstraction. Submittal date: 04/04/2000. Submit to RPC URN-0563

151713 MO0004SPASOL10.002. Radionuclide Solubility Limits. Submittal date: 04/24/2000.

151768 MO0006SPAPVE03.001. Preliminary Volcanic Eruption Biosphere Dose Conversion Factors. Submittal date: 06/15/2000. Submit to RPC URN-0565

151712 MO0007RIB00091.000. Defense High Level Waste Glass Degradation. Submittal date: 07/26/2000.

151812 MO0008SPATHS03.001. Thermal-Hydrological Sensitivity Calculations for Various Ventilation Times, Lineal Heat Loading, and Infiltration Rates in Support of the Report CAL-EBS-HS-000003. Submittal date: 08/24/2000. Submit to RPC URN-0657

153039 MO0009MWDPEN01.009. Pena Blanca Natural Analogue Modeling of the Nopal I Uranium Deposit. Submittal date: 09/15/2000. Submit to RPC URN-0668

152884 MO0010MWDSUP04.010. Supporting Data for Abstraction of Models of Stress Corrosion Cracking of Drip Shield and Waste Package Outer Barrier and Hydrogen Induced Corrosion of Drip Shield. ANL-EBS-PA-000004 REV 00 ICN 01. Submittal date: 10/25/2000. Submit to RPC URN-0646

153127 MO0010MWDWAP01.009. WAPDEG models for tspa-sr. ---*.gsm files are goldsim 6.04.007/ WAPDEG 4.0 inputs and outputs---*.jnb files are sigmaplot 4.0 graphs---files in the runfiles directory are WAPDEG 4.0 input files and dlls---files in the prewap_for_no_backfill directory are for the prewap routine. Submittal date: 10/24/2000. URN-0723

152605 MO0010SPASIL02.002. Silica Adjusted General Corrosion Rates of Alloy 22 and Titanium Grade 7. Submittal date: 10/10/2000.

139569 MO9911SPACDP37.001. In-Package Chemistry Abstraction for Co-Disposal Packages. Submittal date: 11/24/1999.

148596 MO9912SPAPAI29.002. PA Initial Abstraction of THC Model Chemical Boundary Conditions. Submittal date: 01/11/2000.

147818 SN0001T0801500.001. Calculation Tables for the Number of Waste Packages Hit by Igneous Intrusion. Submittal date: 01/21/2000.

147198 SN0001T0872799.006. In-Drift Thermodynamic Environment and Percolation Flux. Submittal date: 01/27/2000.

146931 SN0002T0571599.002. Uncertainty Distributions for Stochastic Parameters. Submittal date: 02/28/2000.

149556 SN0003T0503100.001. Weighting Factors for Low, Middle and High Climate Infiltration Rate Maps. Submittal date: 03/20/2000.

151021 SN0003T0810599.010. Revised Average Radionuclide Activities for Commercial Spent Nuclear Fuel (CSNF) and Co-Disposal Waste Packages for Total System Performance Assessment-Site Recommendation (TSPA-SR) and Final Environmental Impact Statement (TSPA-FEIS). Submittal date: 03/15/2000.

149288 SN0004T0501600.004. Updated Results of the Base Case Saturated Zone (SZ) Flow and Transport Model. Submittal date: 04/10/2000.

151515 SN0004T0501600.005. Updated Input Files to the Base Case Saturated Zone (SZ) Flow and Transport Model for TSPA Abstractions. Submittal date: 04/10/2000.

149254 SN0004T0571599.004. Uncertainty Distributions for Stochastic Parameters Revision to Include New U Sorption Coefficients in the Alluvium and Supporting Electronic Files. Submittal date: 04/10/2000.

150856 SN0006T0502900.002. Updated Igneous Consequence Data for Total System Performance Assessment-Site Recommendation (TSPA-SR). Submittal date: 06/15/2000.

146900 SN9908T0581699.001. Files to Support 1-D Comparison Between FEHM Particle Tracking and T2R3D Advective-Dispersive Transport Simulations Along SD-9. Submittal date: 08/16/1999.

132867 SN9908T0581999.001. Recharge and Lateral Groundwater Flow Boundary Conditions for the Saturated Zone (SZ) Site-Scale Flow and Transport Model. Submittal date: 08/19/1999.

108437 SN9908T0872799.004. Tabulated In-Drift Geometric and Thermal Properties Used in Drift-Scale Models for TSPA-SR (Total System Performance Assessment-Site Recommendation). Submittal date: 08/30/1999.

126110 SN9910T0581699.002. Post-Processed Flow Fields for RIP: Developed Data from AMR U0125 (Abstract Flow Fields for RIP). Submittal date: 10/15/1999.

146902 SN9912T0511599.002. Revised Seepage Abstraction Results for TSPA-SR (Total System Performance Assessment-Site Recommendation). Submittal date: 12/15/1999.

136370 SN9912T0512299.002. Annual Surface Soil Removal Estimates for Amargosa Valley Soils. Submittal date: 12/09/1999.

143657 SNT05070198001.001. Three-Dimensional Rock Property Models for FY98. Submittal date: 07/30/1998.

7.4 OUTPUT DATA

151716 MO0007MWDTSP01.002. TSPA SR, REV 00B, Case SR00 049NM5 Base Case; Nominal Scenario; No Backfill; 300 Realizations; 100,000 Years. Submittal date: 07/19/2000. Submit to RPC URN-0566

151706 MO0007MWDTSP01.003. TSPA SR, REV 00B1, CASE SR00 091NM5 Base Case; Nominal Scenario; No Backfill; 300 Realizations; 100,000 Years. Submittal date: 07/19/2000. Submit to RPC URN-0567

152184 MO0008MWDBARRI.000. Barrier Sensitivity Cases TSPA SR, REV 00B Nominal Scenario; 100,000 Years. Submittal date: 08/31/2000. Submit to RPC URN-0578

152186 MO0008MWDHUMAN.000. Human Intrusion Cases TSPA SR, REV 00B Human Intrusion Scenario, No Backfill. Submittal date: 08/31/2000. Submit to RPC URN-0576

152185 MO0008MWDIGNEO.000. Igneous Sensitivity Cases TSPA SR, REV 00B Igneous Scenario; No Backfill, 300 Realization, 100,000 Years. Submittal date: 08/31/2000. Submit to RPC URN-0577

151720 MO0008MWDIM501.006. TSPA SR, REV 00B, Case SR00_016IM5.---Base Case; Igneous Scenario; No Backfill; 300 Realizations; 100,000 Years. Submittal date: 08/15/2000. Submit to RPC URN-0568

152188 MO0008MWDJUVEN.000. Juvenile Failure Cases TSPA SR, REV 00B Nominal Scenario; No Backfill, 100 Realizations, 100,000 Years. Submittal date: 08/31/2000. Submit to RPC URN-0573

152187 MO0008MWDNEUTR.000. RSS4 Neutralization Cases TSPA SR, REV 00B Nominal Scenario; No Backfill, 100 Realizations, 100,000 Years. Submittal date: 08/31/2000. Submit to RPC URN-0575

151714 MO0008MWDNM501.005. TSPA SR, REV 00B, Case SR00_047NM5.---Base Case; Nominal Scenario; No Backfill; 100 Realizations; 100,000 Years. Submittal date: 08/15/2000. Submit to RPC URN-0569

151719 MO0008MWDNM501.007. TSPA SR, REV 00B, Case SR00_161NM5.---EBS Only Run for Chemistry Plots---Base Case; Nominal Scenario; No Backfill; 300 Realizations; 100,000 Years. Submittal date: 08/15/2000. Submit to RPC URN-0570

153123 MO0009MWDIM401.015. TSPA_SR, REV. 00B, CASE SR00_005IM4, BASE CASE; IGNEOUS SCENARIO; NO BACKFILL; 5000 REALIZATIONS; 50,000 YEARS. Submittal date: 09/20/2000.

153132 MO0009MWDNM501.017. TSPA_SR, REV. 00B1, CASE SR00_108NM5, BASE CASE; NOMINAL SCENARIO; NO BACKFILL; 500 REALIZATIONS; 100,000 YEARS. Submittal date: 09/20/2000.

152839 MO0009MWDNM601.018. Million-Year Sensitivity Cases for the Nominal Scenario. TSPA_SR, Rev. 00B, Case SR00_023NM6 and Case SR00_024NM6. Submittal date: 09/20/2000. Submit to RPC

153131 MO0009MWDTSP01.019. Regression analyses and classification tree analyses for TSPA_SR. Submittal date: 09/20/2000.

153128 MO0011MWDMIL01.022. MILLION-YEAR SENSITIVITY CASES FOR THE NOMINAL SCENARIO, SECONDARY PHASES AND LONG-TERM CLIMATE CHANGE, WITH FIXED PU242 BDCF. TSPA_SR MODEL CASES SR00_034NM6 AND SR00_035NM6. Submittal date: 11/06/2000.

153126 MO0011MWDNM601.021. TSPA_SR MODEL CASE SR00_042NM6. GROUNDWATER PROTECTION BASE CASE; NOMINAL SCENARIO; NO BACKFILL; 300 REALIZATIONS; 1,000,000 YEARS. Submittal date: 11/06/2000.

153269 MO0011MWDREG01.001. REGRESSION ANALYSES AND CLASSIFICATION TREE ANALYSES FOR TSPA_SRCR, REV 00, ICN 01. Submittal date: 11/21/2000.

APPENDIX A
GLOSSARY

APPENDIX A

GLOSSARY

The glossary is divided into two sections. Section A.1 is a general glossary of terms used in the TSPA-SR. Section A.2 contains a listing of statistical terms that are used in or are relevant to other statistical terms used in the TSPA-SR. Definitions are written with emphasis on the relationship of the term to the TSPA-SR process and are taken from previous performance assessment documentation, where possible, or from standard reference materials.

Many of the definitions in this Glossary are Yucca Mountain Site Characterization Project specific.

A.1 GENERAL GLOSSARY

This section is a general listing of terms used in the TSPA-SR. Statistical terms are in Section A.2.

Abiotic	Characterized by the absence of living organisms.
Absorbed Dose	The energy absorbed from ionizing radiation per unit mass of irradiated material. Units of absorbed dose are the rad and the gray (Gy).
Abstracted Model	Model that reproduces, or bounds, the essential elements of a more detailed process model and captures uncertainty and variability in what is often, but not always, a simplified or idealized form. See Abstraction.
Abstraction	Distillation of the essential components of a process model into a suitable form for use in a total system performance assessment. The distillation must retain the basic intrinsic form of the process model but does not usually require its original complexity. Model abstraction is usually necessary to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses.
Actinide	A series of chemically similar, mostly synthetic, radioactive elements with atomic numbers from 89 (actinium) through 103 (lawrencium).
Activity	Cumulative curie count. See Radioactivity.
Adsorb	To collect a gas, liquid, or dissolved substance on a surface as a condensed layer.
Adsorbate	A substance that is adsorbed. See Adsorb.

Adsorbent	A substance upon which another substance is adsorbed. See Adsorb.
Adsorption	Transfer of solute mass, such as radionuclides, in groundwater to the solid geologic surfaces with which it comes in contact. The term sorption is sometimes used interchangeably with this term.
Adsorption Isotherm	Relationship of the quantity of an adsorbed component to its quantity in the fluid phase (expressed in concentration) at constant temperature (i.e., under isothermal conditions).
Adsorption Coefficient	See Sorption Coefficient.
Advection	The process in which solutes are transported by the motion of flowing groundwater. Advection in combination with dispersion (hydrodynamic dispersion) controls flux into and out of the elemental volumes of the flow domain in groundwater transport models. The term convection is sometimes used for advection but is not used interchangeably in the TSPA-SR.
Advisory Committee On Nuclear Waste	A committee established under the U.S. Nuclear Regulatory Commission to provide independent reviews of, and advice on, nuclear waste facilities, including application to such facilities of 10 CFR Parts 60 and 61 (disposal of high-level radioactive wastes in geologic repositories and land disposal of radioactive waste) and other applicable regulations and legislative mandates.
Aerobic	Living or active only in the presence of oxygen, as used in reference to bacteria that require oxygen; a condition in which oxygen is present.
Air Mass Fraction	Mass of air divided by the total mass of gas (typically air plus water vapor) in the gas phase. This expression gives a measure of the "dryness" of the gas phase, which is important in waste package corrosion models.
Algorithm	(1) The set of well-defined rules that governs the solution of a problem in a finite number of steps. (2) A mathematical formulation of a model of a physical process.
Alkaline	See pH.
Alloy-22	See Inner Barrier.

Alluvium	Sedimentary material (clay, mud, sand, silt, gravel) deposited by flowing water or by wind.
Alternative	Plausible interpretations or designs based on assumptions other than those used in the base case that could also fit or be applicable based on the available scientific information. When propagated through a quantitative tool such as performance assessment, alternative interpretations can illustrate the significance of the uncertainty in the base case interpretation chosen to represent the repository's probable behavior.
Ambient	(1) Undisturbed, natural conditions such as ambient temperature caused by climate or natural subsurface thermal gradients. (2) Surrounding conditions.
Anaerobic	(1) Living or active only in the absence of oxygen; used in reference to bacteria that do not require oxygen. (2) A condition in which oxygen is absent.
Anionic	An atom or group of atoms having a negative charge.
Anisotropy	The condition in which physical properties vary when measured in different directions or along different axes. For example, in a layered rock section the permeability is often anisotropic in the vertical direction from layer to layer but is isotropic in the horizontal direction within a layer.
Annual Dose	For human exposure scenarios, a measure of an individual's exposure to radiation in a year.
Annual Committed Effective Dose Equivalent	Composed of terms in 40 CFR 191[103644], Subpart B, in which an annual committed effective dose means the committed effective dose caused by 1-year intake from released radionuclides plus the annual effective dose caused by direct radiation from facilities or activities. See Effective Dose Equivalent and Committed Dose Equivalent.
Annual Frequency	Number of occurrences on an annual basis.
Anthropogenic	Alterations of the environment resulting from the presence or activities of humans.
Aqueous	Pertaining to water, such as aqueous phase, aqueous species, or aqueous transport.

Aquifer	A subsurface, saturated rock unit (formation, group of formations, or part of a formation) of sufficient permeability to transmit groundwater and yield usable quantities of water to wells and springs.
Areal Mass Loading	Used in thermal loading calculations, the amount of heavy metal (usually expressed in metric tons of uranium or equivalent) emplaced per unit area in the potential repository. This number is 85 metric tons of uranium (MTU) per acre and remains a constant value over time for calculations in which the amount of waste per acre in the potential repository is assumed to remain constant.
AREST-CT Computer Program	A general modeling code that considers both equilibrium and kinetically controlled chemical reactions between solid phases, aqueous solutions, and gas under flowing conditions.
Average Individual	An individual representative of the lifestyle in the Amargosa Valley with regard to eating, drinking, and other activities that may be relevant in a human exposure scenario as determined by a survey of Amargosa Valley residents by TSPA-SR researchers.
Backfill	The general fill that is placed in the excavated areas of the underground facility. If used, the backfill for the potential repository may be tuff or other material.
Background Radiation	Radiation arising from natural radioactive material always present in the environment, including solar and cosmic radiation, radon gas, soil and rocks, and the human body.
Basalt	A dark, fine-grained igneous rock originating from a lava flow or minor intrusion, composed mainly of plagioclase clinopyroxene and sometimes olivine, and often displaying a columnar structure.
Base Case	The sequence of anticipated conditions expected to occur in and around the potential repository, without the inclusion of unlikely or unanticipated features, events, or processes. The components that contribute to the base case model are intended to encompass this probable behavior of the potential repository, based on the range of uncertainty for the various parameters and conceptual models used in constructing the base case. In this sense the term is synonymous with nominal case. Base case is also used as a general modeling term to describe a case against which other cases using a different set of assumptions or inputs is compared. Thus, it is possible to have a base case analysis of both nominal and disruptive scenarios.

Base Case Model	A computer model that represents an assessment of the most likely range of behavior for the overall potential repository system and is a combination of the most likely ranges of behavior for the various component models, processes, and associated parameters.
Biosphere	The ecosystem of the earth and the living organisms inhabiting it.
Biosphere Dose Conversion Factor	A multiplier used in converting a radionuclide concentration at the geosphere/biosphere interface into a dose that a human would experience for all pathways considered, with units expressed in terms of annual dose (i.e., the effective dose equivalent) per unit concentration. Depends on the radionuclide(s), pathway(s), climate, and other factors. A key assumption is that the dose is a linear function of concentration at the geosphere/biosphere interface.
Boiling Regime	One of two divisions (the other being the cooling regime) used to delineate the reactions between the gas, water, and minerals in the rock that occur as the system heats and boiling of the pore water occurs through time.
Borehole	A hole drilled from the surface for purposes of collecting information about an area's geology or hydrology. Sometimes referred to as a drillhole or well bore.
Borosilicate Glass	High-level radioactive waste matrix material in which boron takes the place of the lime used in ordinary glass mixtures.
Boundary Condition	For a model, the establishment of a set condition (set value), often at the geometric edge of the model, for a given variable. An example is using a specified groundwater flux from infiltration as a boundary condition for a flow model.
Breach	An opening in the waste package caused by gradual degradation of the outer and inner barriers that allows the waste to be exposed, and possibly released, to the external environment.
Breakthrough	The time at which the concentration of a substance, usually in groundwater, arrives at a particular point of interest after having been tracked as it moves through space.

Breakthrough Curve	A means of describing transport of radionuclides along a geosphere pathway by constructing a curve that is a cumulative probability distribution. The breakthrough curve calculation includes the effects of all flow modes, flow in rock matrix, flow in fractures, and retardation and determines the expected proportion of the radionuclide mass that has traveled the pathway at any specified time.
Buoyant Convection	Fluid movement, typically in the gas phase, in response to a density gradient in a gravitational field. An example is the rising of air when it becomes less dense because of heating followed by its subsequent fall when it cools and becomes denser.
Burnup	A measure of nuclear-reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission or as the amount of energy produced per unit weight of fuel.
Calcite	A crystalline mineral composed of calcium carbonate ($CaCO_3$).
Calibration	(1) The process of comparing the conditions, processes, and parameter values used in a model against actual data points or interpolations (e.g., contour maps) from measurements at or close to the site to ensure that the model is compatible with "reality" to the extent feasible. (2) For tools used for field or lab measurements, the process of taking instrument readings on standards known to produce a certain response to check the accuracy and precision of the instrument.
Canister	The structure surrounding the waste (e.g., high-level radioactive waste immobilized in glass rods) that facilitates handling, storage, transportation, and/or disposal. A metal receptacle with the following purpose: (1) a pour mold for solidified high-level radioactive waste, and (2) for spent nuclear fuel, structural support for loose rods, non-fuel components, or containment of radionuclides during postclosure operations.
Capillarity	(1) A phenomenon that results from the force of mutual attraction (cohesion) between water molecules in conjunction with the force of molecular attraction (adhesion) between water and different solid materials. (2) A means by which water will rise in small diameter tubes and, in combination with the effects of gravity, a means of water movement in the unsaturated zone.
Capillary Barrier	A contact in the unsaturated zone between a geologic unit containing relatively small-diameter openings and a unit containing relatively large-diameter openings across which water does not flow.

Capillary Force	In the unsaturated zone, the forces acting on moisture that can be attributed to the attraction between rock grain, or matrix, surfaces and water.
Capillary Pressure	The difference in a fluid pressure at a given point between a nonwetting phase such as air and a wetting phase such as water.
Capillary Suction	A condition in unsaturated rocks in which the attraction of fluids to particle surfaces is stronger than the force of gravity on the fluid.
Carbon Steel	A steel that is tough but malleable and contains a small percentage of carbon. The inner barrier of waste packages is composed of carbon steel.
Carbonate	Any compound formed by the reaction of carbonic acid with either a metal or an organic compound. Any compound containing the carbonate ion.
Carbonation	A chemical process involving the change of concrete and cement into a carbonate.
Carboniferous	Producing, containing, or pertaining to carbon or coal.
Cationic	An atom or group of atoms having a positive charge.
Center For Nuclear Waste Regulatory Analyses	A federally funded research and development center in San Antonio, Texas, sponsored by the Nuclear Regulatory Commission to provide the Nuclear Regulatory Commission with technical assistance for the repository program.
Ceramic Coating	A layer of ceramic material such as alumina that has been applied to a metallic product to protect against extremely high temperatures and corrosion.
Cladding	The metallic outer sheath of a fuel element generally made of stainless steel or a zirconium alloy. It is intended to isolate the fuel element from the external environment.
Clay	A rock or mineral fragment of any composition that is smaller than very fine silt grains, having a diameter less than 0.00016 in. (1/256 mm). A clay mineral is one of a complex and loosely defined group of finely crystalline hydrous silicates formed mainly by weathering or alteration of primary silicate minerals. They are characterized by small particle size and their ability to adsorb large amounts of water or ions on the surface of the particles.

Climate	Weather conditions, including temperature, wind velocity, precipitation, and other factors, that prevail in a region.
Climate Proxies	The physical remains of substances that carry the imprint of past climates.
Climate States	Representations of climate conditions.
Code (Computer)	The set of commands used to solve a mathematical model on a computer.
Codisposal	A packaging method for disposal of radioactive waste in which two types of waste, such as commercial spent nuclear fuel and defense high-level radioactive waste, are combined in disposal containers. Codisposal takes advantage of otherwise unused space in disposal containers and is more cost-effective than other methods to limit the reactivity of individual waste packages.
Coefficient of Multiple Determination	See Section A.2 of this glossary.
Colloid	As applied to radionuclide migration, a colloidal system is a group of large molecules or small particles that have at least one dimension with the size range of 10^{-9} to 10^{-6}m that are suspended in a solvent. Naturally occurring colloids in groundwater arise from clay minerals such as smectites and illites. Colloids that are transported in groundwater can be filtered out of the water in small pore spaces or very narrow fractures because of the large size of the colloids.
Colloid-Facilitated, Radionuclide Transport Model	A model that represents the enhanced transport of radionuclides by particles that are colloids.
Commercial Spent Nuclear Fuel	Commercial nuclear fuel rods that have been removed from reactor use.
Committed Dose Equivalent	The dose equivalent that is committed to specific organs or tissues that will be received from an intake of radioactive material by an individual during the 50 years following the intake.
Committed Effective Dose Equivalent	The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.

Complementary Cumulative Distribution Function	See Section A.2 of this glossary.
Component Models	The 9 process models that are run separately and then combined for running in the TSPA-SR GoldSim computer model.
Concentration Gradient	For a substance dissolved in a solute, the change in concentration of the substance over a distance.
Conceptual Model	A set of qualitative assumptions used to describe a system or subsystem for a given purpose. Assumptions for the model should be compatible with one another and fit the existing data within the context of the given purpose of the model.
Conduction	Transport of heat in static groundwater, controlled by the thermal conductivity of the geologic formation and the contained groundwater and described by a linear law relating heat flux to temperature gradient.
Confidence	See Section A.2 of this glossary.
Confidence Interval	See Section A.2 of this glossary.
Consequence	A measurable outcome of an event or process that, when combined with the probability of occurrence, gives risk.
Conservative Assumption	(1) An assumption that has the effect of maximizing the calculated amount of radionuclides released from the hypothetical repository to the accessible environment. (2) An assumption that uses uncertain inputs and does not attempt to include any potentially beneficial effects.
Conservative Tracer	Substances with no retardation effect. See Tracer.
Continuous Random Variable	See Section A.2 of this glossary.
Continuum Model	A model that represents fluid flow through numerous individual fractures and matrix blocks by approximating them as continuous flow fields.

Convection	(1) Thermally driven groundwater flow or a heat-transfer mechanism for a gas phase. The bulk motion of a flowing fluid (gas or liquid) in the presence of a gravitational field, caused by temperature differences that, in turn, cause different areas of the fluid to have different densities (e.g., warmer is less dense). (2) One of the processes that moves solutes in groundwater. See Transport.
Convolution Integral Method	(1) A computational method used to calculate the radionuclide concentration in the saturated zone as it changes with time. (2) The abstraction method for the saturated zone flow and transport component model of the TSPA-SR GoldSim computer model.
Cooling Regime	One of two divisions (the other being the boiling regime) used to delineate the reactions between the gas, water, and minerals in the rock, which occur as the system cools after heating and boiling of the pore water occurs through time.
Correlation Coefficient	See Section A.2 of this glossary.
Corrosion	The process of dissolving or wearing away gradually, especially by chemical action.
Corrosion Model (for inner barrier and outer barrier)	A model that includes the time histories of first and subsequent pit and patch penetrations for the waste package layers.
Corrosion Resistant Material	A material that develops a protective film on its surface, creating a high resistance to corrosion. This material, usually the nickel-base alloy, Alloy-22, is used as the outer barrier of the two-layer waste-disposal container.
Coupling	The ability in a performance assessment to assemble separate analyses so that information can be passed among them to develop an overall analysis of system performance.
Covariance	See Section A.2 of this glossary.
Crevice Corrosion	A type of localized corrosion that forms in splits or cracks.
Critical Event	See Criticality.

Critical Group	With regard to annual dose, the maximally exposed individuals. A group of members of the public whose exposure is reasonably homogeneous and includes individuals receiving the highest dose. The individuals making up the critical group may change with changes in source term and pathway.
Criticality	(1) A condition that would require the original waste form, which is part of the waste package, to be exposed to degradation followed by conditions that would allow concentration of sufficient nuclear fuel, the presence of neutron moderators, the absence of neutron absorbers, and favorable geometry. (2) The condition in which nuclear fuel sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.
Critical Population	See Critical Group.
Cumulative Distribution Function	See Section A.2 of this glossary.
Cumulative Probability	See Section A.2 of this glossary.
Cumulative Release	The sum of the radionuclide curies released over a certain period at a specific location.
Curie	A unit of radioactivity equal to 37 billion disintegrations per second.
Darcy's Law	Used in hydrology to describe fluid flow in a porous medium. Darcy's Law states that the fluid velocity is directly proportional to the hydraulic gradient between the two locations.
Data	Facts or figures measured or derived from site characteristics or standard references from which conclusions may be drawn. Parameters that have been derived from raw data are sometimes, themselves, considered to be data.
Decay	See Radioactive Decay.
Deep Percolation	Precipitation moving downward, below the plant-root zone, toward storage in subsurface strata.

Defense in Depth	The term used to describe the property of a system of multiple barriers to mitigate uncertainties in conditions, processes, and events by employing barriers that are redundant and independent, such that failure in any one barrier does not result in failure of the entire system.
Defense Spent Nuclear Fuel	See DOE Spent Nuclear Fuel.
Department of Energy, U.S. (DOE)	A Cabinet-level agency of the United States federal government charged with the responsibilities of energy security, national security, and environmental quality.
Design Concept	As mentioned in the Energy and Water Development Appropriations Act, consists of the subsurface repository layout, the engineered barrier segments, and the waste package.
Desorption	A physical or chemical process by which a substance that has been adsorbed or absorbed by a liquid or solid material is removed from the material.
Deterministic	A single calculation using only a single value for each of the model parameters. A deterministic system is governed by definite rules of evolution leading to cause and effect relationships and predictability. Deterministic calculations do not account for uncertainty in the physical relationships or parameter values.
Diffusion	(1) The spreading or dissemination of a substance. (2) The gradual mixing of the molecules of two or more substances due to random thermal motion.
Diffusive Transport	Movement of solutes due to their concentration gradient. The process in which substances carried in groundwater move through the subsurface by means of diffusion because of a concentration gradient.
Diffusivity	A measure of the rate of heat diffusion. It varies with the nature of the involved atoms, the structure, and changes in temperature.
Dike	A tabular body of igneous rock that cuts across the structure of adjacent rocks or cuts massive rocks. Most dikes are caused by the intrusion of magma. Some dikes occur as columnar structures.
Dimensionality	Modeling in one, two, or three dimensions.

Dimensionality Abstraction	An abstraction in which there is a change in the dimensions of a problem, such as from three dimensional to two dimensional, for modeling purposes. This is done either to simplify the problem or reduce the computational requirements of the problem to implement modeling results in a more efficient or usable form.
Discrete Heat Source	An attribute of drift-scale thermal hydrology models in which the model includes a representation of heat output for discrete waste packages with varying heat outputs depending on the type and amount of waste in the package.
Dispersion (Hydrodynamic Dispersion)	(1) The tendency of a solute (substance dissolved in groundwater) to spread out from the path it is expected to follow if only the bulk motion of the flowing fluid (defection) moved it. The tortuous path the solute follows through openings (pores and fractures) causes part of the dispersion effect in the rock. (2) The macroscopic outcome of the actual movement of individual solute particles through a porous medium. Dispersion causes dilution of solutes, including radionuclides, in groundwater and is usually an important mechanism for spreading contaminants in low flow velocity situations.
Disposal Container	The container barriers or shells, spacing structures or baskets, shielding integral to the container, packing contained within the container, and other absorbent materials designed to be placed internal to the container or immediately surrounding the disposal container (i.e., attached to the outer surface of the container). The disposal container is designed to contain spent nuclear fuel and high-level radioactive waste, but exists only until the outer lid weld is complete and accepted. The disposal container does not include the waste form or the encasing containers or canisters (e.g., high-level radioactive waste pour canisters, DOE spent nuclear fuel codisposal canisters, multi-purpose canisters of spent nuclear fuel, etc.).
Dissolution	Change from a solid to a liquid state. Dissolving a substance in a solvent.
Distribution	See Section A.2 of this glossary.
Distribution Frequency	See Section A.2 of this glossary.
Disturbed Performance	Refers to the behavior of the system if perturbed by disruptive events such as human intrusion or natural phenomena such as volcanism, or nuclear criticality. This is as used in a description of scenario classes, scenarios, or features, events, or processes making up scenarios.

Disruptive Event	An unexpected event that, in the case of the potential repository, includes human intrusion, volcanic activity, seismic activity, and nuclear criticality. Disruptive events have two possible effects: (1) direct release of radioactivity to the surface or (2) alteration of the nominal behavior or the system. For the purposes of screening features, events, and processes for the total system performance assessment, a disruptive event is defined as an event that has a significant effect on the expected annual dose and that has a probability of occurrence during the period of performance less than 1.0 but greater than the cutoff of $10^{-4}/10^{4}$ year defined by the NRC at proposed 10 CFR 63.114(d) (64 FR 8640 [101680]).
Disruptive Event Scenario Class	The scenario, or set of related scenarios, that describes the behavior of the system if perturbed by disruptive events. The disruptive scenarios contain all disruptive features, events, and processes that have been retained for analysis.
Domain (Model)	(1) The set of elements that a mathematical model describes. (2) Individual process areas, such as the unsaturated zone flow domain.
DOE Spent Nuclear Fuel	Radioactive waste created by defense activities that consists of over 250 different types of spent nuclear fuel and is expected to contribute 2,333 metric tons of heavy metal (MTHM) to the total potential repository. The major contributor to this waste form is the N-reactor fuel currently stored at the Hanford Site..
Dose	The amount of radioactive energy that passes the exchange boundaries of an organism (e.g., skin and mucous membranes) and is taken into living tissues. Dose arises from a combination of the energy imparted by the radiation and the absorption efficiency of the affected organism or tissues. It is expressed in terms of units of the radiation taken in, the body weight or mass impacted, and the time over which the dose occurs or the impact is measured.
Dose Conversion Factor	(1) Any factor used to change an environmental measurement to dose in the appropriate units. (2) The multipliers that convert an amount of radionuclides ingested or inhaled to an estimate of dose.
Dose Equivalent	The product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. See also Effective Dose Equivalent and Total Effective Dose Equivalent.
Dose Rate	An organism's exposure to radiation over time.

Downgradient	An area toward which water will tend to flow as the result of several factors. The most important factor is the elevation of water levels in wells in that area relative to other areas. The downgradient is the direction in which contaminants released from the potential repository at Yucca Mountain and migrating in the saturated zone might be expected to move. Based on current understanding of the hydraulic gradient below Yucca Mountain, downgradient is toward the south to southeast of the potential repository location in the area within about 5 km.
Drift	From mining terminology, a horizontal underground passage. The nearly horizontal underground passageways from the shaft(s) to the alcoves and rooms. Includes excavations for emplacement (emplacement drifts) and access (access mains).
Drift Scale	The scale of an emplacement drift, or approximately 5 m in diameter.
Drip Shield	A sheet of impermeable material placed above the waste package to prevent seepage water from directly contacting the waste packages.
Dripping Condition	Assumed for a certain fraction of the waste packages based on water seepage into a drift. The following set of assumptions apply: (1) a small number of the waste packages will be emplaced in drifts with fractures that periodically drip water, and water may drip on a certain fraction of these packages after emplacement; (2) if water drips onto a waste package, it is 100 percent wet from the dripping; and (3) the dripping rate, frequency of drip periods, and water chemistry (especially pH and chloride concentration) will contribute significantly to waste package degradation.
Dual Permeability Conceptual Model	A conceptual model of groundwater flow in which fractures and rock matrix are represented as separate, interacting continua, with no assumption of pressure equilibrium between fractures and rock matrix. This concept allows modeling groundwater flow as occurring mostly in the fractures, with less flow in the rock matrix depending on the degree of connection between the rock matrix and fractures and the capillary pressure gradient. The dual permeability model is one of the conceptual models for groundwater and heat flow for fractured, porous media.
Dual Permeability/Weeps Model	A dual-permeability approximation of the Weeps Model. Also see Dual Permeability Conceptual Model and Weeps Model.

Edge Effects	Conditions at the edges of the potential repository that are cooler and wetter because heat dissipates more quickly than at the center of the repository.
Effective Dose Equivalent	The sum of the products of the dose equivalent to the organ or tissue and the weighting factors applicable to each of the body organs or tissues that are irradiated.
Effective Porosity	The fraction of a given medium's porosity available for fluid flow and/or solute storage, as in the saturated zone.
Electric Power Research Institute	A nonprofit organization that serves as a research and development consortium serving the entire power industry, from power generation to delivery, to end use products and services. This group has performed an independent performance assessment on the Yucca Mountain site.
Elicitation	See Expert Elicitation.
El Niño	A complex set of changes in the water temperature in the Eastern Pacific equatorial region, producing a warm current. This occurs annually to some degree between October and February, but in some years intensifies and causes unusual storms and destruction of marine life.
Empirical Model	A model whose reliability is based on observation and/or experimental evidence and is not necessarily supported by any established theory or law. Validity or applicability of such an empirical model is normally limited to situations that lie within the range of the data that were used to develop the model.
Emplacement Drift	See Drift.
Energy Policy Act of 1992	Comprehensive energy legislation enacted in 1992. Section 801 of the Act directs the U.S. Environmental Protection Agency (EPA) to contract with the National Academy of Sciences to provide "findings and recommendations on reasonable standards that would govern the long-term performance of a repository at the Yucca Mountain site." The EPA Administrator is to promulgate public health and safety standards after the receipt of the findings and recommendations of the National Academy of Sciences, and these shall be the only standards applicable to the Yucca Mountain site.

Engineered Barrier Segments	As mentioned in the 1997 Energy and Water Development Appropriations Act, include (1) the invert and pedestal systems to support the waste package, (2) any packing or backfill materials that may be used within the drift, and (3) any drip shield that may be placed over or around the waste package.
Engineered Barrier System	The waste packages and the underground facility. The designed, or engineered, components of the disposal system and the waste package.
Engineered Barrier System Transport Model	A computer model that includes the key processes: (1) in-drift thermal hydrology and geochemistry, (2) degradation of the drip shield (if used), (3) degradation of the waste package and cladding, (4) alteration and dissolution of the waste form, (5) degradation of the invert, (6) mobilization of the radionuclides in the waste form, and (7) transport of radionuclides in the drift.
Enrichment	The percentage of the fuel matrix that is fissile.
Environmental Impact Statement (EIS)	A detailed written statement to support a decision to proceed with major Federal actions affecting the quality of the human environment. This is required by the National Environmental Policy Act of 1969 [103924]. The environmental impact statement describes:

> ...the environmental impact of the proposed action; any adverse environmental effects which cannot be avoided should the proposal be implemented; alternatives to the proposed action (although the Nuclear Waste Policy Act, as amended, precludes consideration of certain alternatives); the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented.

Preparation of an environmental impact statement requires a public process that includes public meetings, reviews, and comments, as well as agency responses to the public comments.

Environmental Protection Agency (EPA), U.S.	The agency charged by the Nuclear Waste Policy Act of 1982, and subsequently by the Energy Policy Act of 1992, with promulgating generally applicable standards for protection of the general environment. The potential repository at Yucca Mountain is overseen by this agency.
Equilibrium	The state of a chemical system in which the phases do not undergo any spontaneous change in properties or proportions with time, a dynamic balance.
Equilibrium Batch Reactor	A concept describing the conditions in a computer model cell in which the value of any given parameter is homogeneous and in equilibrium throughout the cell area. Used when referring to concentration conditions within an individual cell during modeling of engineered barrier transport.
Equivalent Continuum Model	A conceptual model of groundwater and heat flow that is also called a composite porosity model. Key assumptions are that the temperatures and capillary pressures in the rock matrix and fractures are equal. Therefore, the fractures and matrix can be treated as a single composite material, and the hydraulic properties are a combined effect of both fracture and matrix properties.
Evapotranspiration	The combined processes of evaporation and plant transpiration that remove water from the soil and return it to the air.
Event Tree	A structurally tree-like diagram that is useful in representing sequences of events and their possible outcomes. Each node, or branching point, represents an event, such as volcanic activity, and each branch from that node represents one of its possible outcomes. Each branch can continue to branch many times. Each possible pathway along the tree, from beginning to end of a given line of branching, represents a specific scenario.
Events	(1) Occurrences that have a specific starting time and, usually, a duration shorter than the time being simulated in a model. (2) Uncertain occurrences that take place within a short time relative to the time frame of the model. For the purposes of screening features, events, and processes for the total system performance assessment, an event is defined to be a natural or anthropogenic phenomenon that has a potential to affect disposal system performance and that occurs during an interval that is short compared to the period of performance.
Expected Behavior	The nominal behavior of the potential repository system and the geologic barrier in the absence of disruptive events.

Expected Value	See Section A.2 of this glossary.
Expected Value Realization	The single realization derived by sampling all uncertain input parameters in the component models at the expected values of their ranges.
Expert Elicitation	A formal process through which expert judgment is obtained.
Exploratory Studies Facility	An underground laboratory at Yucca Mountain that includes a 7.9-km (4.9-mile) main loop (tunnel), a 2.8-km (1.75-mile) cross-drift, and a research alcove system constructed for performing underground studies during site characterization. The data collected will contribute toward determining the suitability of the Yucca Mountain site. Some or all of the Exploratory Studies Facility may eventually be incorporated into the potential repository.
External Criticality	A condition in which a critical configuration of fissile material occurs after this material is released from the waste packages. See also Criticality.
Far-Field	With reference to processes, those occurring at the scale of the mountain. The area of the geosphere and biosphere far enough away from the potential repository that, when numerically modeled, represents releases from the repository as a homogeneous, single-source effect.
Fast Path	Localized unsaturated zone flow pathways that might have high advective velocities. Fast paths move water, carrying radionuclides, through the unsaturated zone more quickly than if movement were predominantly through the pores of the rock matrix. Fractures are potential fast paths.
Fault (Geologic)	A fracture in rock along which movement of one side relative to the other has occurred.
Features	Physical, chemical, thermal, or temporal characteristics of the site or potential repository system. For the purposes of screening features, events, and processes for the total system performance assessment, a feature is defined to be an object, structure, or condition that has a potential to affect disposal system performance.
FEHM Computer Code	The Finite Element Heat and Mass transfer computer code that is a process model for unsaturated flow and transport.

Fick's Law	The mass of solute diffusing is proportional to the concentration gradient when a solute in water moves from an area of greater concentration toward an area of lesser concentration by molecular diffusion.
Film Flow	Movement of water as a thin film along a surface.
Finite Difference Computer Code	A commonly used numerical method for solving flow problems. An approximating technique in which algebraic equations are used for approximating the partial differential equations that comprise mathematical models in order to produce a form of the problem that can be solved on a computer. For this type of approximation, the real world area being modeled is formed into a grid with cubical or rectangular blocks. Values for parameters, such as head, are computed at the grid nodes with the same value also being the average for the area surrounding the node.
Finite Element Computer Code	A commonly used numerical method for solving flow problems. An approximating technique in which algebraic equations are used for approximating the partial differential equations that comprise mathematical models in order to produce a form of the problem that can be solved on a computer. For this type of approximation, the real world area being modeled is formed into a grid with irregularly shaped blocks. This method provides an advantage in handling irregularly shaped boundaries, internal features such as faults, and simulation of point sources (of contamination), seepage faces, and moving water table elevations. Values for parameters are frequently calculated at nodes for convenience, but are defined everywhere in the blocks by means of interpolation functions.
Fissile	Sometimes used as a synonym for fissionable (see Fission). Fissile material can undergo fission with neutrons of any energy, including thermal, or slow, neutrons. The three primary materials in this category are uranium-233, uranium-235, and plutonium-239. Fissionable nuclides require fast neutrons to undergo fissions.
Fissile Material	See Fissile.
Fission	The splitting of a nucleus into at least two other nuclei, resulting in the release of two or three neutrons and a relatively large amount of energy.
Fission Products	A complex mixture of nuclides produced by the process of fission that includes radioactive (and some nonradioactive nuclides) as well as the daughter products of the radioactive decay of these nuclides, which can result in more than 200 isotopes.

Flow	The movement of a fluid such as air or water. Flow and transport are groundwater processes that can move potential contaminants; it usually means flow based on Darcy's law.
Flow Pathway	The subsurface course that a water molecule or solute (including radio nuclides) would follow in a given groundwater velocity field governed principally by the hydraulic gradient.
Flux	The rate of transfer of fluid, particles, or energy passing through a unit area per unit time. For water, also known as specific discharge.
Fractures	Breaks in rocks caused by the stresses that cause folding and faulting. A fracture along which there has been displacement of the sides relative to one another is called a fault. A fracture along which no appreciable movement has occurred is called a joint. Fractures may act as fast paths for groundwater movement.
Fracture Aperture	(1) The space that separates the sides of a fracture. (2) The measured width of the space separating the sides of a fracture.
Fracture Permeability	The capacity of a rock to transmit fluid that is related to fractures in the rock.
Fracture-Matrix Exchange Coefficient	(1) A multiplier used in unsaturated groundwater flow simulations that alters the geometric conductance between fracture and matrix elements to account for reduced wetting and contact area. (2) A coefficient that assists in capturing the effect of groundwater being distributed unevenly over fracture surfaces as it moves through fractured rock.
Frequency Distribution	See Section A.2 of this glossary.
Fuel Assembly	A number of fuel rods held together by plates and separated by spacers, used in a reactor. This assembly is sometimes called a fuel bundle.
Fuel Matrix	The physical form and composition of the substance that holds the fissile material.
Fugacity	A parameter that measures the chemical potential of a real gas in the same way that partial pressure measures the free energy of an ideal gas.

Galvanic	Pertaining to an electrochemical process in which electron flow is produced between two dissimilar metals when they are immersed in an electrolyte solution and placed in contact or are electrically connected. The electron flow results from the difference in electrical potential of the metals.
Galvanic Corrosion	Electrochemical corrosion (eating into a substance) caused by the flow of electricity that occurs when two dissimilar metals, with differing electrical potentials, are near each other in the presence of a conductor such as water with solutes in it.
Gaseous Diffusion	The selective transfer of gas by molecular diffusion through microporous barriers. Used to refer to the mechanism for movement of gas through concrete and rock and for movement of gas out of the waste package by means not involving water.
GENII	A deterministic computer software code that evaluates dose from the migration of radionuclides introduced into the accessible environment, or biosphere, that may eventually affect humans through ingestion, inhalation, or direct radiation. It is used to develop biosphere dose conversion factors.
GENII-S	A quasi-stochastic computer software code that can create distributions and sample them and is run in conjunction with GENII for biosphere modeling.
Geochemical	The distribution and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere, and the circulation of the elements in nature on the basis of their properties.
Geochemistry	The study of the abundance of the elements and atomic species (isotopes) in the earth. Geochemistry, or geochemical study looks at systems related to chemicals arising from natural rock, soil, soil processes such as microbe activity, and gases, especially as they interact with man-made materials from the potential repository system. In the broad sense, all parts of geology that involve chemical changes.
Geologic-Framework Model	A nonmathematical model of the geologic system.
Geologic Repository	A system for disposing of radioactive waste in excavated geologic media, including surface and subsurface areas of operation, and the adjacent part of the geologic setting that provides isolation of the radioactive waste.

Geologic Time	The period of time over which the earth has existed. The time scale over which geologic processes produce change. In general discussion, the term geologic time implies very long periods of time such as tens of thousands of years, hundreds of thousands of years, or millions of years.
Geosphere	The combination of the earth's rock, water, and air layers (spheres).
Glass	See High-Level Radioactive Waste Glass.
Goethite	An iron oxide mineral that is yellowish, reddish, or brownish black. It is the most common constituent of many forms of natural rust or of limonite.
Gradient	The change in value of a quantity per unit distance in a specified direction.
Groundwater	Water contained in pores or fractures in either the unsaturated or saturated zones below ground level.
Groundwater Flux	The rate of groundwater flow through a unit area of the aquifer. Means the same as specific discharge.
Groundwater Travel Time	The time required for a unit volume of groundwater to travel between two locations. The travel time is the length of the flow path divided by the velocity, where velocity is the average groundwater flux divided by the effective porosity along the flow path. If discrete segments of the flow path have different hydrologic properties, the total travel time will be the sum of the travel times for each discrete segment.
Handling Container	The container in which the fuel matrix and cladding are placed. If the waste form is solidified, this is called a pour container. In some cases, this is the only container for storage, handling, and transportation prior to disposal.
Heavy Metal	All uranium, plutonium, and thorium used in a nuclear reactor.
Herbivore	An organism that feeds on plants, especially an animal whose diet is exclusively plants.

Heterogeneity	The condition of being composed of parts or elements of different kinds. A condition in which the value of a parameter such as porosity, which is an attribute of an entity of interest such as the tuff rock containing the potential repository, varies over the space an entity occupies, such as the area around the repository, or with the passage of time.
High-Level Radioactive Waste	(1) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing, and any solid material derived from such liquid waste that contains fission products in sufficient concentrations. (2) Other highly radioactive material that the U.S. Nuclear Regulatory Commission, consistent with existing law, determines by rule requires permanent isolation.
High-Level Waste	See High-Level Radioactive Waste.
High-Level Radioactive Waste Glass	The waste form of defense high-level radioactive waste in which the radioactive waste is mixed with borosilicate glass.
Histogram	See Section A.2 of this glossary.
Homogeneous	Consisting of or composed of similar elements or ingredients.
Host Rock	The rock unit in which the potential repository will be located. For the potential repository at Yucca Mountain, the host rock would be the middle portion of the of the Topopah Spring tuff formation of the Paintbrush Group. See also tuff.
Hydraulic Conductivity	A measure of the ability to transmit water through a permeable medium. A number that describes the rate at which water can move through a permeable medium. The hydraulic conductivity depends on the size and arrangement of water-transmitting openings such as pores and fractures, the dynamic characteristics of the water such as density and viscosity, and the strength of the gravitational field.
Hydraulic Gradient	The change in the height of water levels with respect to the distance between two locations.
Hydrodynamic Dispersion	See Dispersion.
Hydrogeology	A study that encompasses the interrelationships of geologic materials and processes involving water.

Hydrologic	Pertaining to the properties, distribution, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.
Hydrology	(1) The study of water characteristics, especially the movement of water. (2) The study of water, involving aspects of geology, oceanography, and meteorology.
Hydrostratigraphy	A stratigraphic classification of layered rocks based on rock characteristics and the hydrologic, or water-conducting, properties of the units.
Human Intrusion	The inadvertent disturbance of a disposal system by humans that could result in release of radioactive waste. The regulations require that performance assessments consider the possibility of human intrusion.
Igneous	(1) A type of rock that has formed from a molten, or partially molten, material. (2) A type of activity related to the formation and movement of molten rock either in subsurface (plutonic) or on the surface (volcanic).
Imbibition	The absorption of a fluid, usually water, by porous rock (or other porous material) under the force of capillary attraction and without pressure.
Incolloy 625	Under past reference design specifications, the corrosion-resistant inner layer of the two-layer metallic disposal container..
Infiltration	The process of water entering the soil at the ground surface and the ensuing movement downward when the water input at the soil surface is adequate. Infiltration becomes percolation when water has moved below the depth at which it can be removed (to return to the atmosphere) by evaporation or evapotranspiration.
Infiltration Flux	Volumetric infiltration rate per unit area.
Infiltration Rate	See Infiltration Flux.
Inner Barrier	An inner layer of the two-layer metallic disposal container.
Inner Canisters	High-level radioactive waste canisters placed within the overpack.
In Situ	In its natural position or place. The phrase distinguishes in-place experiments, conducted in the field or underground facility, from those conducted in the laboratory.

Integral-Finite-Difference Computer Code	A commonly used numerical method for solving flow problems. An approximating technique in which algebraic equations are used for approximating the partial differential equations that comprise mathematical models in order to produce a form of the problem that can be solved on a computer. Similar in capability to a finite element code in that it can handle irregularly shaped areas well. See Finite Element Computer Code.
Inventory	The amount of radioactive elements in a fuel, usually stated in curies per metric ton of heavy metal. Also termed radionuclide inventory.
Invert	A construction associated with the precast concrete structure for the purpose of providing a level drift floor and enabling transporting and support of the waste package.
Ion	(1) An atom that contains excess electrons or is deficient in electrons, causing it to be chemically active. (2) An electron not associated with a nucleus.
Ionizing Radiation	(1) Alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. (2) Any radiation capable of displacing electrons from an atom or molecule, thereby producing ions.
Ionic Strength	A measure of the level of electrical force in an electrolytic solution.
Irradiated Fuel	Burned fuel. See also Burnup.
Isothermal	Pertaining to constant temperature.
Isotope	One of two or more atomic nuclei with the same number of protons (i.e., the same atomic number) but with a different number of neutrons (i.e., a different atomic weight). For example, ^{235}U and ^{238}U are both isotopes of uranium.
Isotropy	The condition wherein all significant physical properties are equal when measured in any direction or along any axes. See also Anisotropy.
Iterative	Conditions or results that are repeated in an analysis. The processes in which analysts rerun calculations or refine models as new data are gathered or new insights occur.
ITOUGH2 Computer Code	A computer code that estimates hydrogeologic model parameters for the numerical simulator TOUGH2.

J-13 Water	The groundwater taken from Wellbore J13. The chemical composition of this water is used as the standard for Yucca Mountain ambient groundwater composition for modeling purposes.
Joint	A fracture in rock, usually more or less vertical to bedding, along which no appreciable movement has occurred.
Juvenile Failure	Premature failure of a waste package because of material imperfections or damage by rockfall during emplacement.
Key Technical Issues	Issues important for assessing the long-term safety of a potential Yucca Mountain repository, as defined by the U.S. Nuclear Regulatory Commission (NRC). The issues are (1) Support Revision of the U.S. Environmental Protection Agency Standard/NRC Rule Making; (2) Total System Performance Assessment and Technical Integration; (3) Igneous Activity; (4) Unsaturated and Saturated Flow Under Isothermal Conditions; (5) Thermal Effects on Flow; (6) Container Life and Source Term; (7) Structural Deformation and Seismicity; (8) Evolution of Near-Field Environment; (9) Radionuclide Transport; (10) Repository Design and Thermal Mechanical Effects.
Kinetic	Of or due to motion.
Latin Hypercube Sampling	A sampling technique that divides the cumulative distribution function into intervals of equal probability and then samples from each interval.
License Application	An application to the Nuclear Regulatory Commission for a license to construct a repository.
Line Loading Repository Design	A waste emplacement design in which waste containers are spaced very closely along the drift, with emplacement drifts relatively far apart.
Lithophysal	Pertaining to tuff units with lithophysae, voids having concentric shells of finely crystalline alkali feldspar, quartz, and other materials that were formed due to entrapped gas that later escaped.
Lithosphere	The earth's crust, as distinguished from the atmosphere or hydrosphere, and as distinguished from the deeper portion of the earth underlying the crust.

Localized Corrosion	A type of corrosion induced by local variations in electrochemical potential on a microscale over small regions. Variations in electrochemical potential may be caused by localized irregularities in the structure and composition of usually protective passive films on metal surfaces and in the electrolyte composition of the solution that contacts the metal. See Pitting Corrosion and Crevice Corrosion.
Log Normal Distribution	A distribution of a random variable X such that the natural logarithm of X is normally distributed.
Lookup Table	A multidimensional table containing columns of data representing relationships between parameters in the table. A lookup table is a convenient way to represent and implement functional relationships between parameters considered in the model.
Longitudinal Dispersion	(1) Dispersion of a solute moving in groundwater in the same direction as the groundwater flow path. (2) Spreading of a solute in the direction of bulk flow.
Magma	Molten or partially molten rock material that is naturally occurring and is generated within the earth.
Mass Balance	The procedure of accounting for conservation of mass, such as the mass of radionuclides released from waste packages, in real world processes or in models of real world processes.
Mathematical Model	A mathematical description of a conceptual model.
Matrix	Tuff rock material and its pore space exclusive of fractures. As applied to Yucca Mountain tuff, the groundmass of an igneous rock that contains larger crystals.
Matrix Diffusion	As used in TSPA-SR conceptual models, the process by which molecular or ionic solutes, such as radionuclides in groundwater, move from areas of higher concentration to areas of lower concentration. This movement is through the pore spaces of the rock material as opposed to movement through the fractures.
Matrix Permeability	The capacity of the matrix to transmit fluid.
Mean (Arithmetic)	See Section A.2 of this glossary.

Mechanistic Analysis	An analysis of processes that is based on the well-established fundamentals of the processes considered, such as: thermodynamics, reaction kinetics, mass transfer laws, heat transfer laws, etc. This is as opposed to empirical analysis, which is based on a model that has been developed from the numerical value of data taken from tests or measurements of the model.
Median	A value such that half of the observations are less than that value and half are greater than the value.
Meteorological	Of, or relating to meteorology, or to weather and other atmospheric phenomena.
Metric Ton Heavy Metal (MTHM)	A metric ton is a unit of mass equal to 1,000 kg (2,205 lb.). Heavy metals are those with atomic masses greater than 230. Examples include thorium, uranium, plutonium, and neptunium. When used in the Civilian Radioactive Waste Management Program, the term usually pertains to heavy metals in spent nuclear fuel in scientific text. In this document, MTHM is equal to MTU (metric tons of uranium).
Metric Ton of Uranium (MTU)	A metric ton, which is 1,000 kg, or 2,205 lb., of uranium in scientific text.
Microbe	An organism too small to be viewed with the unaided eye. Examples of microbes are bacteria, protozoa, and some fungi and algae.
Microbially Influenced Corrosion	Corrosion of the waste package that is enhanced by the activity of microbes.
Microbiologically Influenced Corrosion	See Microbially Influenced Corrosion.
Migration	Radionuclide movement from one location to another within the engineered barrier system or the environment.
Mild Steel	See Carbon Steel.
Mineral Assemblage	Minerals that compose a rock, especially an igneous or metamorphic rock. The term includes the different kinds and relative abundance of minerals but excludes the texture and fabric of the rock.
Mineralogical	Of or relating to the chemical and physical properties of minerals, their occurrence, and classification.

Mobile Radionuclides	Radionuclides that can move within a water system with little or no retardation.
Mobilization	The process of breaking down the waste form and releasing radionuclides. After its initial mobilization a radionuclide can be removed from transport by being precipitated or adsorbed and later become remobilized in a cycle of changes that can be repeated many times.
Model	A depiction of a system, phenomenon, or process including any hypotheses required to describe the system or explain the phenomenon or process.
Molal	Of a solution, containing one mole of solute per one kilogram of solvent.
Mole	The fundamental metric unit used to measure the amount of a substance. Avogadro's number of particles (6.023×10^{23}).
Monte Carlo Uncertainty Analysis	See Section A.2 of this glossary.
Mountain Scale	(1) Similar to far-field for processes that are related to the area of the geosphere and biosphere far enough away from the potential repository that, when numerically modeled, show that releases from the repository are represented as a homogeneous, single source term. The effects of individual, small-scale components such as individual waste packages are not modeled because they are considerably smaller than the scale of the model. (2) A scale of hundreds of meters, or even kilometers, as opposed to tens of meters.
National Academy of Sciences	A congressionally chartered, private, nonprofit, self-perpetuating organization of scientists devoted to the expansion of science and its use for the general welfare. This organization is mandated to advise the Federal government on scientific and technical matters. Section 801 of the Energy Policy Act of 1992 directed the U.S. Environmental Protection Agency to contract with the National Academy of Sciences to provide, "findings and recommendations on reasonable standards that would govern the long-term performance of a potential repository at the Yucca Mountain site."
National Research Council	The working arm of the National Academy of Sciences and the National Academy of Engineering that carries out most of the studies done on behalf of the academies. Most of the studies are done in response to specific questions presented by federal agencies or Congress.

Natural Analogs	Natural geologic systems that parallel situations that can develop in man-made systems, in which the formation and transport of minerals over hundreds of thousands and millions of years can be studied directly. An example of natural analog is the natural reactor studied at the Oklo uranium deposit in Gabon, Africa, which can be used as a source of analog data for conceptual models of criticality.
Near-Field	The area and conditions within the potential repository including the drifts and waste packages and the rock immediately surrounding the drifts. The region around the potential repository where the natural hydrogeologic system has been significantly impacted by the excavation of the repository and the emplacement of waste.
Near-Field Geochemical Environment Model	A model that focuses on major-element geochemistry within the potential emplacement drifts. The boundary of the model domain is defined as the drift wall. This model includes coupling to thermohydrologic processes.
Net Infiltration	The water that has infiltrated down from the soil zone or exposed rock surface to a depth below which it cannot be removed by evapotranspiration. The amount of water that is net infiltration is the total infiltration at the surface minus water lost to evaporation and plant transpiration.
Neutron Absorber	A material such as boron or gadolinium that is placed in a radioactive waste package and that absorbs neutrons to reduce ionizing radiation and to help reduce the likelihood of criticality.
Node	A junction point in a network.
Nominal Case	The case, or conceptual model, representing the expected conditions of the disposal system as perturbed only by the presence of the potential repository, in the absence of disruptive events.
Nominal Conditions	The site conditions, including features and processes, which are expected, based on current site knowledge.
Nominal Behavior	(1) Expected behavior of the system as perturbed only by the presence of the potential repository. (2) Behavior of the system in the absence of disruptive events.

Nominal Scenario Class	The scenario, or set of related scenarios, that describes the expected or nominal behavior of the system as perturbed only by the presence of the potential repository. The nominal scenarios contain all expected features, events, and processes that have been retained for analysis.
Nominal Features, Events, and Processes	Those features, events, and processes expected, given the site conditions as described from current site characterization information.
Nonequilibrium Thermodynamics	The study of heat flow systems that have not stabilized (i.e., are not in equilibrium).
Nuclear Chain Reaction	A process in which some of the neutrons released in one fission event cause other fissions.
U.S. Nuclear Regulatory Commission	Promulgates technical regulations that are consistent with standards established by the U.S. Environmental Protection Agency and considers license applications from the U.S. Department of Energy for a potential repository. It determines, with reasonable assurance, whether EPA standards can be met. It also has the continuing regulatory responsibility to oversee repository operation. U.S. Nuclear Regulatory Commission was formed by the Atomic Energy Commission with the Energy Reorganization Act of 1974 [100213].
Nuclear Regulatory Commission Radioactive Waste Program Annual Progress Report	A status report made each fiscal year that documents the technical work performed on 10 key technical issues that are most important to performance of the potential geologic repository at Yucca Mountain.
Nuclear Waste Policy Act (42 U.S.C. 10101 et seq.)	The federal statute enacted in 1982 that established the Office of Civilian Radioactive Waste Management and defined its mission to develop a federal system for the management and geologic disposal of commercial spent nuclear fuel and other high-level radioactive wastes. The Act also: (i) specified other federal responsibilities for nuclear waste management, (ii) established the Nuclear Waste Fund to cover the cost of geologic disposal, (iii) authorized interim storage under certain circumstances, and (iv) defined interactions between Federal agencies and the states, local governments, and Indian tribes. The act was substantially amended in 1987 and 1992.

Nuclear Waste Policy Amendments Act of 1987	Legislation that amended the Nuclear Waste Policy Act to: (i) limit repository site characterization activities to Yucca Mountain, Nevada, (ii) establish the Office of the Nuclear Waste Negotiator to seek a state or Indian tribe willing to host a repository or monitored retrievable storage facility, (iii) create the Nuclear Waste Technical Review Board, and (iv) increase state and local government participation in the waste management program.
Nuclear Waste Technical Review Board	An independent body established within the executive branch, created by the Nuclear Waste Policy Amendments Act of 1987 to evaluate the technical and scientific validity of activities undertaken by the U.S. Department of Energy, including site characterization activities and activities relating to the packaging or transportation of high-level radioactive waste or spent nuclear fuel. Members of this Board are appointed by the President from a list composed by the National Academy of Sciences.
NUFT Computer Code	A computer code that simulates three-dimensional flow of groundwater, heat, and contaminants in unsaturated and saturated porous and fractured media. It is named for Non-isothermal Unsaturated Flow and Transport and is used for drift scale, thermal-hydrologic calculations.
Numerical Model	An approximate representation of a mathematical model that is constructed using a numerical description method, such as finite volumes, finite differences, or finite elements. A numerical model is typically represented by a series of program statements that are executed on a computer.
Office of Civilian Radioactive Waste Management	A U.S. Department of Energy office created by the Nuclear Waste Policy Act of 1982 to implement the responsibilities assigned by the Act.
One-Dimensional Model	A model that represents physical conditions and/or processes by a vertical column composed of a stack of single grid cells or by a horizontal row of single grid cells.
Order of Magnitude	A range of numbers extending from some value to 10 times that value.
Outer Barrier	The outer layer of the two-layer metallic disposal container. It consists of carbon steel, which is a corrosion allowance material.
Outer Barrier and Inner Barrier Corrosion Models	See Corrosion Models.

Outer Barrier Corrosion Model	See Outer Barrier and Inner Barrier Corrosion Models.
Overburden	Geologic material of any nature, consolidated or unconsolidated, that overlies a deposit of useful materials. As used by the Yucca Mountain Site Characterization Project, this is geologic material overlying the mined repository horizon.
Oxidation	(1) A chemical reaction, such as the rusting of iron, that increases the oxygen content of a substance. (2) A reaction in which the valence of an element or compound is increased as a result of losing electrons.
Oxidation State	For an ion, expressed as a positive or negative number representing the ionic or effective charge.
Oxidize	(1) To increase the oxygen content of a substance. (2) To increase the valence of an element or compound as a result of losing electrons.
Paleoclimates	The climate of a past interval of geologic time.
Parameter	Data, or values, that are input to computer codes for a TSPA calculation.
Passive Institutional Control	From 40 CFR 191, methods of preserving information about the location, design, and contents of the potential repository system. These include permanent markers placed around the disposal site area, public records and archives, government ownership, and regulations controlling use of land.
Patch	For corrosion modeling, one of two geometries for an opening in a waste package layer created by corrosion (the other geometry is a pit). A patch is generally wider than it is deep.
Pathway	A potential route by which radionuclides might reach the accessible environment and pose a threat to humans.
Peer Review Panel	A panel of individuals independent of those who performed the TSPA-SR but who have technical expertise at least equivalent to those who performed the original work who produce a documented critical review of the work.
Percentile	See Section A.2 of this glossary.

Perched Water	A saturated condition that is not continuous with the water table, because there is an impervious or semipervious layer underlying the perched zone or a fault zone that creates a barrier to water movement and perches water.
Percolation	The passage of a liquid through a porous substance. In rock or soil it is the movement of water through the interstices and pores under hydrostatic pressure and the influence of gravity. The downward or lateral flow of water that becomes net infiltration in the unsaturated zone.
Percolation Flux	Volumetric percolation rate per unit area. The flux anywhere below the root zone of plants and is no longer susceptible to removal back into the atmosphere by evapotranspiration.
Percolation Rate	See percolation flux.
Performance Assessment	An analysis that predicts the behavior of a system or system component under a given set of constant and/or transient conditions. Performance assessments will include estimates of the effects of uncertainties in data and modeling. See Total System Performance Assessment.
Permeability	In general terms, the capacity of a medium such as rock, sediment, or soil to transmit liquid or gas. Permeability depends on the substance transmitted (oil, air, water, etc.) and on the size and shape of the pores, joints, and fractures in the medium and the manner in which they are interconnected. "Hydraulic conductivity" has replaced "permeability" in technical discussions relating to groundwater. See also Relative Permeability.
pH	A number indicating the acidity or alkalinity of a solution. A pH of 7 indicates a neutral solution. Lower pH values indicate more acidic solutions while higher pH values indicate alkaline solutions.
Phase	A physically distinct portion of matter, such as the aqueous, gas, or solid phase.
Phase Equilibria	The relationships between phases of a substance undergoing a phase change, such as from solid to liquid, under various conditions of temperature and pressure.
Phase Stability	A measure of the ability of matter to remain in a given phase.
Pit	For corrosion modeling, one of two geometries for an opening in a waste package layer created by corrosion (the other geometry is a patch). A pit is generally deeper than it is wide.

Pitting Corrosion	A type of localized corrosion that forms in indentations called pits.
Pitting Factor	A factor that is used to measure local variations of general, or uniform, corrosion penetration from corrosion allowance materials such as carbon steel.
Playa	The shallow central basin of a desert plain in which water gathers after a rain and then evaporates.
Plume	A measurable discharge of a contaminant, such as radionuclides, from a point of origin. The contaminants are usually moving in groundwater, and the plume may be defined by chemical concentration gradients.
Pluvial	(1) In climatology, relating to former periods of abundant rains, especially in reference to glacial periods. (2) In geology, said of a geologic episode, change, process, deposit, or feature caused by the action or effects of rain.
Point Loading Thermal Design	An emplacement drift design in which commercial spent nuclear fuel waste packages are spaced away from each other along the drift using emplacement drift spacing similar to the commercial spent nuclear fuel-package spacing.
Pore Fluid	The water and any material it is carrying that exist in the pore spaces of the rock matrix.
Pore Waters	Interstitial water, or subsurface water in the pores in rock or soil.
Porosity	The ratio of openings, or voids, to the total volume of a soil or rock expressed as a decimal fraction or as a percentage. See also Effective Porosity.
Pour Canister	A metallic canister into which high-level radioactive waste mixed with molten glass-making materials is poured. The material cools and solidifies in the pour canister.
Precipitate	A substance that separates as solid particles from a liquid as a result of physical or chemical changes.
Precipitation	(1) The process of depositing a substance from a solution, by the action of gravity or by a chemical reaction. (2) Any form of water particles, such as frozen water in snow or ice crystals, or liquid water in raindrops or drizzle, that fall from clouds in the atmosphere and reaches the earth's surface. (3) An amount of water that has fallen at a given point over a specified period of time, measured by a rain gauge.

Probabilistic	(1) Based on or subject to probability. (2) Involving a variate, such as temperature or porosity. At each instance of time, the variate may take on any of the values of a specified set with a certain probability. Data from a probabilistic process is an ordered set of observations, each of which is one item from a probability distribution.
Probabilistic Risk Assessment	(1) A systematic process of identifying and quantifying the consequences of scenarios that could cause a release of radioactive materials to the environment. (2) Using predictable behavior to define the performance of natural, geologic, human, and engineered systems for thousands of years into the future using probability distributions (see Section A.2 of this glossary).
Probability	See Section A.2 of this glossary.
Probability Density Function	See Section A.2 of this glossary.
Probability Distribution	See Section A.2 of this glossary.
Probability Model	A model that quantifies uncertainties in the model parameters and predicts the likelihood of the scenarios used for the model.
Probable Behavior	A combination of the concept of predicted future behavior of the various system components with the uncertainty associated with the prediction.
Process Model	A depiction or representation of a process along with any hypotheses required to describe or to explain the process.
Processes	Phenomena and activities that have gradual, continuous interactions with the system being modeled. For the purposes of screening features, events, and processes for the total system performance assessment, a process is defined as a natural or anthropogenic phenomenon that has a potential to affect disposal system performance and that operates during all or a significant part of the period of performance.
Pyroclastic Flow	A flow of detrital volcanic materials that have been explosively ejected from a volcanic vent. The flow is generally a dense cloud of incandescent volcanic glass, in a semimolten or viscous state, that solidifies into rock. The rock that results is chiefly a fine-grained rhyolitic tuff formed of glass shards that may be welded or nonwelded.

Quantitative	A variable that is expressed numerically.
Quality Factor	The modifying factor that is used to derive dose equivalent from absorbed dose.
Quasi-Static Thermodynamic Processes	Reversible processes resulting in a change to system or body.
Quasi-Transient	Describing the diffusive mass transport model. This means the solution used in the model incorporates steady-state diffusive mass transfer through the perforations of the failed waste container. This is combined with transient mass transfer through the spherical shell of the invert surrounding the waste container. The quasi-transient mass transfer model is used to calculate diffusive release of radionuclides at the engineered barrier system edge.
Rad	The unit of measure for the absorbed dose of radiation. Rad is short for radiation absorbed dose.
Radiation	Ionizing radiation.
Radioactive Decay	The process in which one radionuclide spontaneously transforms into one or more different radionuclides, which are called daughter radionuclides.
Radioactivity	The property possessed by some elements (i.e., uranium) of spontaneously emitting alpha, beta, or gamma rays by the disintegration of atomic nuclei.
Radiocolloid	Colloids, or colloidal systems, containing radionuclides.
Radiolysis	Chemical decomposition by the action of radiation.
Radionuclide	Radioactive type of atom with an unstable nucleus that spontaneously decays, usually emitting ionizing radiation in the process. Radioactive elements characterized by their atomic mass and number.
Random Variable	See Section A.2 of this glossary.
Range (Statistics)	See Section A.2 of this glossary.
REACT Computer Code	The reaction mass transfer code.
Reaction Kinetics	The study of the rates and mechanisms of chemical reactions.
Reaction Rate	The rate at which a chemical reaction takes place.

Realization | A complete calculation using a randomly selected value. Many of these calculations are done in a Monte Carlo analysis.

Realization	A complete calculation using a randomly selected value. Many of these calculations are done in a Monte Carlo analysis.
Recharge	The movement of water from the unsaturated zone to the saturated zone.
Reducing Conditions	With regard to criticality, the important aspect of reducing conditions is that they reduce the oxidation state of materials (deoxidize), and the material that is reduced becomes less soluble. Radionuclides being transported in groundwater can precipitate out and collect in an area of reducing conditions. With regard to corrosion, reducing conditions slow corrosion, because oxygen is less available, or not available, to combine with the iron and form rust.
Reference Person	With regard to dose, a hypothetical collection of human physical and physiological characteristics arrived at by international consensus. This collection may be used by researchers to relate biological damage to a stimulus such as radiation exposure. The reference adult person lives 20 km (12 miles) from Yucca Mountain and will be defined using a survey of the existing population.
Reflux Water	Water that is vaporized near waste packages, migrates to cooler areas, condenses, and then flows back toward the waste packages.
Regression Analysis	See Section A.2 of this glossary.
Relative Permeability	The permeability of rock material to a given substance compared to the absolute (total) permeability of the rock. The term is usually used to signify the permeability to one fluid when two or more fluids are present in the rock.
rem	The unit of a dose equivalent from ionizing radiation to the human body. It is used to measure the amount of radiation to which a person has been exposed) (rem means roentgen equivalent man).
Repository Layout	The host rock, depth, and areal extent of the repository facility, drift size and spacing, mechanical support system, thermal load, and ventilation system used during the operational phase of the facility. This is as mentioned in the Energy and Water Development Appropriations Act of 1997.

Repository Safety Strategy	A document used to assist management in prioritizing testing and analysis activities to focus on the most important issues in postclosure safety. Identification of the important issues allows resource use (e.g., sampling and testing activities) to be focused on gathering information that will reduce the uncertainty in parameters and processes related to the key issues. Key elements of the document include the following: (1) Limited water contacting waste packages (2) Long waste package lifetime (3) Low rate of release of radionuclides from breached waste packages (4) Radionuclide concentration reduction during transport from the waste packages.
Retardation	Slowing or stopping of radionuclide movement in groundwater by mechanisms that include sorption of radionuclides, diffusion into rock matrix pores and microfractures, and trapping of large colloidal molecules in small pore spaces or dead ends of microfractures.
RIP Computer Program	RIP is an initialism for repository integration program, the executive TSPA "driver" program. An integrating software code into which simplified analytical expressions, or callable subroutines describing the behavior of the different components, can be placed. RIP sequentially advances through time while keeping track of the changes in environments and the fate of the radioactive constituents within the engineered and natural barriers.
Risk	The probability that an undesirable event will occur multiplied by the consequences of the undesirable event.
Risk Assessment	An evaluation of potential consequences or hazards that might be the outcome of an action. This assessment focuses on potential negative impacts on human health or the environment.
Rock Matrix	See Matrix.
Salt Deposit Effect	(1) Potential buildup of salt scales on the waste package surface from water dripping onto the waste package while its surface is at elevated temperatures. (2) The development of potentially aggressive conditions to the waste package corrosion degradation under and around the salt deposits by providing a wetter environment than the surroundings and causing concentration of aggressive species in the local salt solution.

Saturated Zone	The region below the water table where rock pores and fractures are completely saturated with groundwater.
Scenario	A well-defined, connected sequence of features, events, and processes that can be thought of as an outline of a possible future condition of the potential repository system. Scenarios can be undisturbed, in which case the performance would be the expected, or nominal, behavior for the system. Scenarios can also be disturbed, if altered by disruptive events such as human intrusion or natural phenomena such as volcanism, or nuclear criticality.
Scenario Class	A set of related scenarios that share sufficient similarities that they can usefully be aggregated for the purposes of screening or analysis. The number and breadth of scenario classes depends on the resolution at which scenarios have been defined. Coarsely defined scenarios result in fewer, broad scenario classes, whereas narrowly defined scenarios result in many narrow scenario classes. Scenario classes (and scenarios) should be aggregated at the coarsest level at which a technically sound argument can be made, while still maintaining adequate detail for the purposes of the analysis.
Secondary Phase	Occurs when spent nuclear fuel is contacted by water and dissolves, forming uranyl minerals. The major secondary phase minerals are schoepite, uranophane, Na-boltwoodite, and soddyite.
Seepage	The inflow of groundwater moving in fractures or pore spaces of permeable rock to an open space in the rock such as a drift. Specifically, the amount of percolation flux that enters the drift in a given time period. An important factor in waste package degradation and mobilization and migration of radionuclides out of the potential repository.
Seepage Fraction	The fraction of the total number of waste packages that is contacted by drips from seepage into the drifts.
Seismic	Pertaining to, characteristic of, or produced by earthquakes or earth vibrations.
Sensitivity Study (Analysis)	An analytic or numerical technique for examining the effects of varying specified parameters when a model run is performed. Shows the effects that changes in various parameters have on model outcomes and can illustrate which parameters have a greater impact on the predicted behavior of the system being modeled. Also, called sensitivity analysis because it shows the sensitivity of the consequences (e.g., radionuclide release) to uncertain parameters (e.g., the infiltration rate that results from precipitation).

Simulation	The generation of a sample set by selecting a parameter value from each input distribution and calculating the consequences for the sample set. See also Realization.
Single Heater Test	A field test in the Exploratory Studies Facility that uses a single heated element emplaced directly into Yucca Mountain tuff (Topopah Spring Middle Nonlithophysal hydrogeologic unit). The test is designed to determine the thermal hydrologic responses of the unit to heating.
Site Characterization Plan	The plan that contains the strategy for completing a detailed set of activities that was expected to provide all of the information needed to comprehensively describe the potential repository system. The plan also documented methods for assessing the performance of the total repository system and its individual components. This was published by the U.S. Department of Energy in 1988 with subsequent, ongoing updating.
Site Recommendation	A recommendation by the Secretary of Energy to the President that the Yucca Mountain site be approved for development as the nation's first high-level radioactive waste repository. If the site is determined to be suitable, this recommendation is expected in fiscal year 2001.
Smeared Heat Source	An attribute of mountain-scale thermal hydrology models in which the model handles heat output for waste packages by using the total heat produced by all assemblies in all waste packages, arrives at the entire repository-wide thermal load, and averages the thermal load across the entire repository heat area (~740 acres).
Sorb	To undergo a process of sorption.
Sorption	The binding, on a microscopic scale, of one substance to another. A term that includes both adsorption and absorption. The sorption of dissolved radionuclides onto aquifer solids or waste package materials by means of close-range chemical or physical forces is an important process modeled in this study. Sorption is a function of the chemistry of the radioisotopes, the fluid in which they are carried, and the mineral material they encounter along the flow path.
Sorption Coefficient (K_d)	Coefficient for a term for the various processes by which one substance binds to another.
Source Term	Types and amounts of radionuclides that are the source of a potential release from the potential repository.

Spalling	Flaking off of corrosion products from the metal substrate as it undergoes corrosion. The layer of corroded material thickens. The spalling could be caused by an expansive action of the corrosion products because they occupy a greater volume than the uncorroded metal substrate.
Spatial Variability	A measure of how a property, such as rock permeability, varies in an object such as a rock formation.
Speciation	The existence of the elements, such as radionuclides, in different molecular forms in the aqueous phase.
Spent Nuclear Fuel	Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Spent fuel that has been burned (irradiated) in a reactor to the extent that it no longer makes an efficient contribution to a nuclear chain reaction. This fuel is more radioactive than it was before irradiation, and it is hot. See also Burnup.
Steady-State Modeling	Modeling a system under the assumption that the variables are not changing with time. For example, flow fields can be simulated at a steady state if the boundary conditions, saturations, and fluxes are not changing with time.
Stream Tube	A modeling method used to represent the groundwater flow path from the water table to the biosphere. There are six stream tubes used for saturated zone modeling with one tube associated with, and having the cross-sectional shape of, one of six regions designated at the water table. Each stream tube takes in groundwater flux and radionuclide mass flux data at the water table representing flux from the potential repository that has gone through the unsaturated zone.
Stochastic	Involving a variable (e.g., temperature and porosity) that may take on values of a specified set with a certain probability. Data from a stochastic process is an ordered set of observations, each of which is one item from a probability distribution. Random.
Stochastic Model	A model whose outputs are predictable only in a statistical sense. A given set of model inputs produces outputs that are not the same, but follow statistical patterns.

Stratigraphy	The branch of geology that deals with the definition and interpretation of the rock strata, the conditions of their formation, character, arrangement, sequence, age, distribution, and especially their correlation by the use of fossils and other means of identification.
Stress Corrosion Cracking	A cracking process that requires the simultaneous action of a corrosion mechanism and sustained tensile stress.
Structural Failure	Loss of waste package integrity to contain radionuclides.
Structure	In geology, the arrangement of the parts of the geologic feature or area of interest such as folds or faults. Structural features develop as a result of stresses that cause movements of the earth's crust and result in events such as earthquakes as the crust deforms.
Surface Complexation	The process that describes the formation of complex molecules between the solute in the aqueous phase and the reactive groups on the solid surface, under specific chemical conditions.
Surrogate	Using one thing in place of another. An example is using a single important parameter, radionuclide travel time, as a surrogate for performance when the actual performance measure takes into account the effects of many factors. If the calculated travel time of radionuclides of interest is fast, that implies that performance of the natural and engineered barriers in containing the radionuclides is not as effective as it would be if radionuclide travel time was slow.
Tectonic	Pertaining to geologic forms or effects created by deformation of the earth's crust.
Tectonism	A general term for all movement of the earth's crust produced by tectonic processes.
Temperature Gradient	The rate of change of temperature with distance. When applied to the earth, the term geothermal gradient may be used.
Thermal-Chemical	Relating to thermal chemistry, the chemistry branch that studies heat changes that accompany chemical reactions and changes of state.
Thermal Conduction	The flow of thermal energy through a material. This conduction is affected by the amount of heat energy present, the nature of the heat carrier in the material, and the amount of dissipation.

Thermal-Hydrologic	Of or pertaining to changes in groundwater movement due to the effects of changes in temperature.
Thermal-Hydrologic Processes	Processes that are driven by a combination of thermal and hydrologic factors. These processes include evaporation of water near the potential repository when it is hot and subsequent redistribution of fluids by convection, condensation, and drainage.
Thermal Hydrology	The study of a system that has both thermal and hydrologic processes. A thermal-hydrologic condition, or system, is expected to occur if heat-generating waste packages are placed in the potential repository at Yucca Mountain.
Thermal Loading	The application of heat to a system, usually measured in terms of watt density. The thermal loading for a repository is the watts per acre produced by the radioactive waste in the active disposal area. The spatial density at which waste packages are emplaced within the potential repository as characterized by the areal power density and the areal mass loading.
Thermal Period	The time period in which thermal effects, such as higher temperatures or dried rock, are present in the region surrounding the potential repository.
Thermal Power Per Waste Package	The rate of heat released in watts by a particular waste package type. This will vary with fuel type and age, waste package capacity, and disposal configurations within waste packages.
Thermodynamic	Pertaining to the mechanical action of heat.
Thermodynamic/Kinetic Coefficients	Numbers that represent the rate of heat flow through a porous medium. An example is a coefficient that represents the rate of heat flow in a given type of rock.
Three-Dimensional Model	A three-dimensional representation of physical conditions and/or processes.
Time History	The predicted response of a system, expressed as a function of simulation time.
TOSPAC Computer Code	A computer code that simulates one-dimensional groundwater flow with the transport of decaying contaminants in partially saturated, fractured media.
Total Effective Dose Equivalent	The sum of the deep-dose equivalent, for external exposures, and the committed effective dose equivalent, for internal exposures.

Total System Performance Assessment (TSPA): A risk assessment that quantitatively estimates how the potential Yucca Mountain repository system will perform in the future under the influence of specific features, events, and processes, incorporating uncertainty in the models and data. Its purposes follow:

(1) Provide the basis for predicting system behavior and testing that behavior against safety measures in the form of regulatory standards
(2) Provide the results of TSPA analyses and sensitivity studies
(3) Provide guidance to site characterization and repository design activities
(4) Help prioritize testing and selection of the most effective design options.

TOUGH2 Computer Code: A computer program that simulates three-dimensional flow of groundwater and heat in unsaturated and saturated porous and fractured media. The code name is derived from Transport of Unsaturated Groundwater and Heat.

Toxicity: The ability of a substance to cause damage to cells or tissues of living organisms when the substance is inhaled, ingested, or absorbed by the skin. Acute toxicity is that which occurs over a short term of exposure, and chronic toxicity is that which occurs over a long term of exposure.

Tracer Testing: A procedure in which a soluble substance (tracer) is added to groundwater at one location, and its movement to another location is observed. Tracer testing is a technique by which groundwater flow directions and velocities and other hydrologic properties of rocks can be estimated.

Transient: Describing a variable that is changing with time. This occurs before development of a steady-state condition.

Transient Modeling: Modeling of a system in which the variables are changing through time. Heating of the potential repository by the waste is a transient condition for which transient modeling is done.

Transparency	According to the Nuclear Waste Technical Review Board, the ease of understanding the process by which a study was carried out, which assumptions are driving the results, how they were arrived at, and the rigor of the analyses leading to the results. According to a Peer Review Panel report, transparency "requires ensuring completeness and using a logical structure that facilitates in-depth review of the relevant issues achieved when a reader or reviewer has a clear picture of what was done in the analysis, what the outcome was, and why."
Transpiration	The process in which water enters a plant through its root system, passes through its vascular system, and is released into the atmosphere through openings in its outer covering. It is an important process for removal of water that has infiltrated below the zone where it could be removed by evaporation.
Transport	A process in which substances carried in groundwater move through the subsurface by means of the physical mechanisms of convection, diffusion, and dispersion and the chemical mechanisms of sorption, leaching, precipitation, dissolution, and complexation. Types of transport include advective, diffusive, and colloidal transport.
Transverse Dispersion	Dispersion of a solute moving in groundwater in directions transverse to the direction of the groundwater flow path. This movement may also be called lateral dispersion.
Tritium	A radioactive isotope of hydrogen that can be taken into the body easily, because it is chemically identical to natural hydrogen. Tritium decays by beta emission with a half-life of about 12.5 years.
TSPA Peer Review Panel	See Peer Review Panel.
Tuff	Igneous rock formed from compacted volcanic fragments from pyroclastic (explosively ejected) flows with particles generally smaller than 4 mm (0.16 in.) in diameter. The most abundant type of rock at the Yucca Mountain site.
Tuffaceous	A general term referring to any rock containing tuff.

Two-Dimensional Model	A two-dimensional slice through an entity, such as the earth's crust, usually in the horizontal and vertical directions, on which known features are placed and are used to predict likely features that may exist between points of known data. Mathematically, a model that represents physical conditions and/or processes; this mathematical model is composed of both horizontal rows and vertical columns of grid cells arrayed in L-shaped configurations only one grid cell thick. Also called a cross section.
Uncertainty	A measure of how much a calculated or estimated value, that is used as a reasonable guess or prediction, may vary from the unknown true value.
Undisturbed Performance	Refers to the expected or nominal behavior of the system as perturbed only by the presence of the potential repository. This is as used in the description of scenario classes, scenarios, or features, events, or processes making up scenarios.
Unsaturated Zone	The zone of soil or rock below the ground surface and above the water table in which the pore spaces contain water, air, and other gases. Generally, the water saturation is below 100 percent in this zone, although areally limited perched water bodies (having 100 percent water saturation) may exist in the unsaturated zone. Also called the vadose zone.
Unsaturated Zone Flow	The flow of water in the unsaturated zone by downward percolation and by capillary action.
Unsaturated Zone Radionuclide Transport Model	A computer software code that defines the movement of radionuclides from the edge of the engineered barrier system, through the unsaturated zone, and to the boundary of the saturated zone.
Vadose Zone	See Unsaturated Zone.
Variable	See Section A.2 of this glossary.
Variability (Statistical)	A measure of how a quantity varies over time or space.
Velocity Field	The velocities of fluid flow, gas or liquid, in a region, which are generally depicted by arrows to indicate the direction and magnitude of the velocity.
Vitrified	Pertaining to a type of processed high-level radioactive waste where the waste is mixed with glass-forming chemicals and put through a melting process. The melted mixture is then put into a canister where it becomes a dry "log" of waste in a glassy matrix.

Vitrified Defense High-Level Radioactive Waste	A type of processed defense high-level radioactive waste that has been contained in a glass matrix.
Volcanism	Pertaining to volcanic activity.
WAPDEG Computer Code	A computer software code that was developed to model long-term corrosion degradation of waste disposal containers in the potential repository.
Waste Containment and Isolation Strategy	A document designed to assist the project management in prioritizing testing and analysis activities to focus on the most important issues in postclosure safety. It is also designed to help resolve uncertainty in processes and parameters of greatest significance to long-term performance. The document is still evolving. The key elements include the following: (1) Low groundwater flow amounts through storage area (2) Long-lived waste packaging (3) Cladding on the waste and low water content in waste to slow degradation (4) Engineered systems that promote slow dispersion/migration of radionuclides (5) Natural systems that promote slow dispersion/migration of radionuclides.
Waste Form	A generic term that refers to radioactive materials and any encapsulating or stabilizing matrix.
Waste Package	The waste form and any containers (i.e., disposal container barriers and other canisters), spacing structures or baskets, shielding integral to the container, packing contained within the container, and other absorbent materials immediately surrounding an individual waste container placed internally to the container or attached to the outer surface of the disposal container. The waste package begins its existence when the outer lid welds are complete and accepted.
Waste Package Design Organization	The management and oversight department responsible for the waste package design and testing.
Waste Stream	Input of waste into the potential repository over time.
Water Flux	See Flux.

Water Table	The upper surface of a zone of saturation above which the majority of pore spaces and fracture openings are less than 100 percent saturated with water most of the time (unsaturated zone), and below which the opposite is true (saturated zone).
Weeps Model	A stochastic conceptual model of groundwater flow through fractured rock. The flow is assumed to occur through stochastically generated fracture paths, or weeps, with no interaction occurring between fracture and matrix.
Welded	Fused.
Welded Tuff	A tuff that was deposited under conditions where the particles making up the rock were heated sufficiently to cohere. In contrast to nonwelded tuff, welded tuff is considered to be denser, less porous, and more likely to be fractured (which increases permeability).
Young Spent Fuel, Old Spent Fuel	Terms used to designate groups of commercial spent nuclear fuels by their age since discharge from the power reactor. The young spent nuclear fuels are characterized by higher radiation levels and resulting higher heat outputs than the old spent nuclear fuels.
Yucca Mountain Waste Containment and Isolation Strategy	See Waste Containment and Isolation Strategy.
Zeolites	A large group of hydrous aluminosilicate minerals that act as molecular "sieves" because they can adsorb molecules with which they interact. At Yucca Mountain, they are secondary alteration products in tuff rocks when the rocks are exposed to groundwater and could act to retard the migration of radionuclides by their sieving action.
Zircaloy	An alloy material that may have any of several compositions including zirconium oxide. It is used as a cladding material.

A.2 GLOSSARY OF STATISTICS TERMS

Terms in this section are presented separately from the general glossary in Section A.1 because many of the statistical terms are defined in relation to other statistical terms. The terms are numbered to allow reference from the general glossary in Section A.1.

Coefficient of Multiple Determination	A measure of goodness of fit of a linear-regression model; a value near 1 indicates a good fit, meaning that the model is accounting for most of the uncertainty in the performance measure being analyzed.

Complementary Cumulative Distribution Function	A method of depicting the probability that a performance measure, such as dose, exceeds a given value. For most measures, the higher the value, the lower the probability.
Confidence	In statistics, a measure of how close the estimated value of a random variable is to its true value.
Confidence Interval	An interval that is believed, with a preassigned degree of confidence, to include the particular value of the random variable that is estimated.
Continuous Random Variable	Those variables whose value is determined by taking measurements and that can take <u>any</u> value of an infinite number of possible values within a certain value range. The concentration of radionuclides in water is a continuous random variable and, although ranging from zero to a value limited by the solubility of an individual radionuclide under given conditions, possible outcomes of dissolving a given radionuclide in water cannot be represented by a finite number of discrete values. This type of variable has a probability density function.
Correlation Coefficient	A coefficient (designated r) calculated in the analysis of paired data when neither of the variables can be singled out as of prior importance to the other and the study seeks to analyze their interdependence, as opposed to the dependence of one upon the other. This term is a dimensionless quantity that can be used (with certain reservations) as an absolute measure of the relationship between two variables. Mathematically, for two random variables, the ratio of their covariance to the product of their standard deviations. The correlation coefficient is also a measure of how close a scatter plot of points produced by one variable plotted against the other comes to falling on a straight line drawn through the trend of the points. In a negative correlation between the two variables, larger values of one are associated with smaller values of the other. In a positive correlation, larger values of one variable are associated with larger values of the other.

Covariance	For a pair of random variables, the expected value of the product of the deviations from their respective means. It measures the extent to which two variables vary together and, if the variables are independent, the covariance is zero (so is the correlation coefficient). If large values of one variable are associated with large values of the other, the covariance is positive, while if small values of one are associated with large values of another, the covariance is negative. The covariance is usually calculated to find the correlation coefficient.
Cumulative Distribution	For grouped data, a distribution that shows how many of the values are less than or more than specified values. For random variables, this term is synonymous with distribution function.
Cumulative Distribution Function	For a continuous random variable, a function that quantifies the probability that the variable is no greater than any specified value of interest. The derivative of the cumulative distribution function is the probability density function. The cumulative distribution function is most commonly used to analyze continuous variables when data are not divided into categories (grouping by some qualitative description), and the probability density function is more appropriate when categorical studies of continuous random variables are performed.
Cumulative Probability	The probability that a random variable will have a value equal to, or less than, some specified value.
Dependent Variable	A variable whose value depends on one or more other variables. For example, the value (amount) of body weight is a variable that depends on several independent variables—the amount of calories taken in and the amount of calories burned, as well as genetics and probably other factors. As another example, the thermal load per acre of the potential repository is a dependent variable—it depends on the type, number, and spacing of waste packages emplaced.

Discrete Random Variables	Those variables whose values are finite, or countable in numbers. The number of waste packages of each type is a discrete variable. Discrete random variables have associated with them probability functions that tell the probability that the variable takes on any particular value. For example, in throws of two unbiased dice, the probability that the value of the numbers shown on the dice (a discrete random variable) for any throw will be two is one in 36; the probability function is 1/36.
Distribution	The overall scatter of values for a set of observed data. A term used synonymously with frequency distribution. Distributions have probability structures that are the probability that a given value occurs in the set.
Distribution Frequency	A representation of how values of an outcome or variable are distributed over the range of expected values.
Distribution Function	A function whose values are the probabilities that a random variable assumes a value less than or equal to a specified value. Synonymous with cumulative distribution.
Expected Value	A variable's mean, or average, outcome. The weighted average of the number of possible outcomes, with each outcome being weighted by its probability of occurrence. The mean of a probability distribution of a random variable that one would expect to find in a very large, random sample. The sum of the possible values, each weighted by its probability—the center of the random variable's histogram (frequency distribution).
Frequency Distribution	Formed when data are grouped into classes (or ranges of values within the overall set of values, such as 1 to 5, 5 to 10, 10 to 20, etc.), with the classes listed in a table (or other format) showing the number of data points that occur in each class.
Function (Mathematics)	A quantity that is variable and whose value depends on and varies with the value of another quantity or quantities. Functions show the mathematical relationship between dependent variables and the independent variables upon which the value of the dependent variables depend.

Histogram	A bar graph representation of a frequency distribution having frequency of occurrence as the ordinate (y axis) and classes of values observed in sampling of the variable as the abscissa (x axis). The area of each rectangle in the histogram represents the proportion of observations (relative frequency) that fall in that interval. This is the relative frequency of observations that lie between the two values that form the class boundary. It is not for a single value but is relative frequency of the class interval.
Linear Correlation	The relationship between two or more random variables for which the regression equations are linear.
Linear Regression	A regression where the relationship between the (conditional) mean of a random variable and one or more independent variables can be expressed by the mathematical equation that describes a line. A relationship between two variables such that the dependence of one variable on the other can be described by (the equation of) a straight line.
Linear Stepwise Regression	An analysis designed to determine variables that have the greatest influence on an output value (e.g., peak dose rate) when there are many variables whose input values go into the calculation. In simple terms, a linear regression is performed for a line in a multidimensional space, and the correlation of the values of different variables to the line are examined by performing the calculation multiple times and varying the value of one variable at a time while holding the others constant. This is a stepwise process in which one variable at a time is examined to determine the impact of its influence on the final outcome (peak dose rate, for instance).
Mean (Arithmetic)	For a statistical data set, the sum of the values divided by the number of items in the set. The arithmetic average.
Mode	A measure of location in a data set defined as the value that occurs with the highest frequency. For qualitative data it is the attribute that occurs most frequently. A set of data or a distribution can have more than one mode, or if no two values are alike, no mode. For the distribution of a random variable, the mode is the value for which the probability function or probability density is at the relative maximum.

Monte Carlo (Uncertainty) Analysis
An analytical method that uses random sampling of parameter values available for input into numerical models as a means of approximating the uncertainty in the process being modeled. A Monte Carlo simulation comprises many individual runs of the complete calculation using different values for the parameters of interest as sampled from a probability distribution. A different final outcome for each individual calculation and each individual run of the calculation is called a realization. Each realization is equally likely to occur in the Monte Carlo process.

Percentile
For a large data set where specific values are not repeated extensively, used to indicate where a value lies in relation to the entire group of values. For example, the 25th percentile indicates that about 25 percent of the items are smaller than this value and about 75 percent are larger than this value.

Probability
The relative frequency with which an event occurs in the long run. Statistical probability is about what really happens in the real world and can be verified by observation or sampling. Knowing the exact probability of an event is usually limited by the inability to know, or compile the complete set of, all possible outcomes over time or space.

Probability-Density Function
A frequency distribution such that the bars of a histogram that would represent it are so narrow that their tops would form a smooth curve if connected by a line. The curve is the probability density function. This type of distribution can be made if the number of observations of the value of a continuous random variable increases indefinitely, and the width of the range represented by each class (class interval) becomes smaller and smaller. The area under the density function curve between any two points on the curve, such as x_1 and x_2, represents the probability that the value of the random variable will lie between these two values.

Probability Distribution
The set of outcomes (values) and their corresponding probabilities for a random variable.

Quantile
A value at or below that lies a given fraction (1/5, 30 percent, etc.) of a set of data. Also called fractile.

R^2
A correlation coefficient that quantifies the goodness of fit of a linear regression model to an output value such as peak dose rate. A value of one corresponds to a perfect fit.

R^2 - Loss	The amount of change in fit when a variable is dropped from a linear stepwise regression analysis. For example, look at a linear stepwise regression analysis such that the output (e.g., dose) is calculated using 10 variables and the total R^2 is 0.80 (1 corresponds to a perfect fit). If the analysis is then performed with one of the variables left out and the R^2 is 0.78 (meaning it changed or lost very little), then that variable does not contribute strongly to the fit. If the loss is large such as going from 0.80 to 0.60, then the variable does contribute strongly to the fit. This is a method of showing to which variables the outcome (peak dose) is most sensitive or responsive.
Random Variable	A property that has a numerical description and is determined by the outcome of a random experiment or random sampling. The different values of the random variable have different probabilities of occurrence. Also called variates.
Range (Statistics)	The numerical difference between the highest and lowest value in any series.
Rank Transformation	A type of data transformation used either to reduce the influence of extreme values or to deal with non-linearities in data sets. Data will fit better to a non-linear curve if it is first put into ranks. In ranking, the data values of both input and input data are replaced with the rank of that data value within the data set. The smallest value of a data set is replaced with the number 1, the second smallest is replaced with the number 2, and so forth up to the largest value in the set.
Regression	The relationship between the (conditional) mean of a random variable and one or more independent variables.
Regression Analysis	The analysis of paired data such that one member of the pair is a constant and the other is a random variable. The analysis of a paired dependent variable and the independent variable upon which it depends. For example, the term was first used in a study of the heights of fathers and sons where a regression (or turning back) was observed toward the mean height of the population in the heights of sons whose fathers were taller or shorter than the mean.
Scatter Plot	(1) A set of points arrived at by plotting paired values as points in a plane. (2) A two-dimensional dot plot.

Standard Deviation	(1) For a set of observations or a frequency distribution, the square root of the average of the squared deviations from the mean divided by n-1 (where n is the sample size). (2) The square root of the variance.
Variable	A nonunique property or attribute.
Variance	(1) The square of the standard deviation. (2) The expected squared distance from the population mean of a random variable, sometimes called the population variance.

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APPENDIX B

SUMMARY OF SCREENING DECISION AND BASIS INFORMATION CONTAINED IN REVISION 00 OF THE YUCCA MOUNTAIN PROJECT AND FEATURES, EVENTS, AND PROCESSES DATABASE

APPENDIX B

ACRONYMS AND ABBREVIATIONS

AECL	Atomic Energy of Canada, Ltd.
AMR	Analysis Model Report
Bio	Biosphere
CA	comparison approach
CRIT	Criticality
CSNF	commercial spent nuclear fuel
DE	disruptive events
DOE	U.S. Department of Energy
DSNF	Department of Energy spent nuclear fuel
EBS	engineered barrier system
FEP	feature, event, and process
FFC	far-field criticality
HLW	high-level radioactive waste
HMIP	Her Majesty Inspectorate of Pollution
IDGE	in-drift geochemical environment
IRSR	Issue Resolution Status Report
ISC	in-situ criticality
MLD	master logic diagram
NAGRA	National Cooperative for the Disposal of Radioactive Waste (Nationale Genossenschaft Fur die Lagerung Radioaktiver Abfalle) (Switzerland)
NEA	U.S. Nuclear Energy Agency
NFC	near-field criticality
NFE	near-field environment
NRC	U.S. Nuclear Regulatory Commission
PMR	Process Model Report
RIG	Revised Interim Guidance
SAM	Safety Assessment Management, Ltd.
SKB	Svensk Karnbranslehantering AB (Swedish Nuclear Fuel and Waste Management Co.)
SKI	Swedish Nuclear Power Inspectorate
SYS	system-level

APPENDIX B

ACRONYMS AND ABBREVIATIONS (Continued)

SZ	saturated zone
TBV	to be verified
TH	thermal-hydrology
THC	Thermal-Hydrologic-Chemical
TM	Thermal-Mechanical
TSPA	Total System Performance Assessment
TSPAI	Total System Performance Assessment and Integration
TSPA-SR	Total System Performance Assessment for Site Recommendation
UZ	unsaturated zone
WF	waste form
WF Misc.	waste form degradation – miscellaneous
WF Clad	waste form – cladding
WF Col	waste form – colloids
WIPP	Waste Isolation Pilot Plant
WP	waste package
YMP	Yucca Mountain Project
YSCP	YMP Site Characterization Plan

APPENDIX B

SUMMARY OF SCREENING DECISION AND BASIS INFORMATION CONTAINED IN REVISION 00 OF THE YUCCA MOUNTAIN PROJECT AND FEATURES, EVENTS, AND PROCESSES DATABASE

B.1. INTRODUCTION

Under the provisions of the U.S. Department of Energy's (DOE) Interim Guidance (Dyer 1999 [105655]), a performance assessment is required to demonstrate compliance with the postclosure performance objectives for the Yucca Mountain Project (YMP). Dyer (1999 [105655], Section 102[j]) defines a performance assessment as a systematic analysis that (1) identifies the features, events, and processes (FEPs) that might affect the performance of the potential geologic repository, (2) examines the effects of such FEPs on the performance of the potential geologic repository, and (3) estimates the expected annual dose to a specified receptor group. The performance assessment must also provide the technical basis for inclusion or exclusion of specific FEPs in the performance assessment (Dyer 1999 [105655], Section 114). To address these requirements, the YMP has adopted an approach to selecting scenarios for analysis in the Total System Performance Assessment for the Site Recommendation (TSPA-SR) that is based on the identification and screening of FEPs potentially relevant to the postclosure performance of the potential Yucca Mountain repository (see Section 2.1.1.1 of the main body of this report).

The electronic YMP FEP Database (CRWMS M&O 2000 [150806], Appendix D) catalogs the YMP FEPs and their associated screening information, which are an integral part of the scenario analysis for TSPA-SR. The five-step scenario-analysis approach for TSPA-SR is consistent with the five elements of Subissue 2, Scenario Analysis outlined in the Issue Resolution Status Report (IRSR) Key Technical Issue: Total System Performance Assessment and Integration (TSPAI) (NRC 2000 [149372], Section 4.2). The five steps are:

1. Identification of FEPs
2. Classification of FEPs
3. Screening of FEPs
4. Formation of Scenario Classes
5. Screening of Scenario Classes.

The information in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]) was developed external to the database—no original information or calculations were generated within the database itself. REV00 of the database contains the following information, which specifically addresses the first three steps of the scenario analysis approach (and, correspondingly, the first three elements of TSPAI IRSR Subissue 2):

- **YMP FEP List**–A comprehensive list of FEPs that have the potential to influence repository performance

- **FEP Classifications**–The categorization of FEPs in accordance with a hierarchical organizational structure that groups similar FEPs together and allows for relationships between FEPs to be identified

- **FEP Screening Decisions and Supporting Documentation**–For each FEP, the technical basis for inclusion or exclusion in the TSPA-SR analyses is summarized, as taken from FEP Analysis Model Reports (AMRs).

The information catalogued in the database, specifically the included (screened-in) FEPs, provides the basis for scenario class formation and screening, the final two steps of the scenario analysis approach. However, these two steps (and, correspondingly, the fourth and fifth elements of TSPAI IRSR Subissue 2) are outside the scope of the database and are addressed in Section 2.1 of the main body of this report.

This Appendix discusses the following:

- The origin and development of a comprehensive list of FEPs potentially relevant to the postclosure performance of the repository
- The development and structure of an electronic database capable of storing and retrieving information about the inclusion and (or) exclusion of these FEPs in TSPA-SR
- A summary of the FEPs and their dispositions (i.e., inclusion and [or] exclusion of these FEPs in the TSPA-SR).

The origin and development of the YMP FEP list is described in Section B.2 of this Appendix. The development of the FEP classifications and the organizational structure of the database are described in Section B.3. Section B.4 is an overview of the screening criteria and guidance for exclusion of FEPs from the TSPA-SR. A summary of screening decisions and bases for all primary FEPs is given in Section B.5. Section B.6 discusses the transparency and traceability, comprehensiveness, categorization, and screening of the YMP FEPs relative to the TSPAI IRSR subissues. A brief summary of this Appendix is given in Section B.7.

The FEP screening decisions and supporting documentation (collectively referred to as the screening discussions) provided in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D) and in Section B.5 were taken from FEP AMRs listed in Table B-1. Each FEP AMR was associated with a Process Model Report (PMR) subject area. Each FEP AMR was prepared in accordance with AP-3.10Q [152363], *Analyses and Models,* and provided qualified documentation of the screening decisions for each FEP relevant to the subject area. Technical details of specific screening discussions and screening criteria are documented in the FEP AMRs, not in this Appendix or in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]). However, a general discussion of the nature of the screening discussions is presented in Section B.4.

Table B-1. Features, Events, Processes Analysis Model Reports Contributing Screening Information to the Yucca Mountain Project Features, Events, and Processes Database

PMR Subject Area	FEP AMR Document Identifier	Reference
Unsaturated Zone (UZ) Flow and Transport	ANL-NBS-MD-000001 REV00	CRWMS M&O 2000 [142945]
Saturated Zone (SZ) Flow and Transport	ANL-NBS-MD-000002 REV00	CRWMS M&O 2000 [137359]
Biosphere (Bio)	ANL-MGR-MD-000011 REV00	CRWMS M&O 2000 [142844]
Disruptive Events (DE)	ANL-WIS-MD-000005 REV00	CRWMS M&O 2000 [146681]
Waste Package (WP) Degradation	ANL-EBS-PA-000002 REV00	CRWMS M&O 2000 [146538]
Waste Form (WF) Degradation–Miscellaneous FEPs (WF Misc.)	ANL-WIS-MD-000009 REV00	CRWMS M&O 2000 [146498]
– Cladding FEPs (WF Clad)	ANL-WIS-MD-000008 REV00	CRWMS M&O 2000 [150099]
– Colloid FEPs (WF Col)	ANL-WIS-MD-000012 REV00	CRWMS M&O 2000 [125156]
Near Field Environment (NFE)	ANL-NBS-MD-000004 REV00	CRWMS M&O 2000 [142895]
Engineered Barrier System (EBS) Degradation, Flow, and Transport	ANL-WIS-PA-000002 REV00	CRWMS M&O 2000 [136951]
System-Level (SYS[a]) FEPs	ANL-WIS-MD-000019 REV00B	CRWMS M&O 2000 [152216]
Criticality (CRIT[a]) FEPs	Not available for database REV00	N/A

Source: CRWMS M&O 2000 [150806]

NOTES: [a] Not a PMR
NA = Not Available

The YMP FEP Database REV00 (CRWMS M&O 2000 [150806]) evolved from preliminary versions REV00A, REV00B, and REV00C. The evolution of the database versions leading to REV00 is described in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806], Section 5).

B.2. IDENTIFICATION OF THE YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES LIST

The development of a comprehensive list of FEPs potentially relevant to the postclosure performance of the potential Yucca Mountain repository is an ongoing, iterative process, based on site-specific information, design, and regulations. The list of FEPs catalogued in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D) was developed using the following approach:

- Develop an initial list of general FEPs from other radioactive waste disposal programs.
- Supplement the general list with FEPs from project-specific literature.
- Augment the list through brainstorming and iterative review from CRWMS M&O subject matter experts (e.g., at technical workshops and in technical reports).
- Augment the list with feedback from external sources (e.g., U.S. Nuclear Regulatory Commission (NRC)/DOE Technical Exchange and Appendix 7 Meetings, NRC IRSRs).

This approach combines the bottom-up (i.e., nonsystematic, all-inclusive) identification of an initial FEP list, with a top-down (i.e., systematic) series of reviews.

B.2.1 INTERNATIONAL FEATURES, EVENTS, AND PROCESSES

The YMP FEPs list was initially populated with 1,261 FEPs compiled by other radioactive waste programs. The FEPs were taken from Version 1.0 of an electronic FEP database (SAM n.d. [139333]) maintained by the U.S. Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development. The NEA database contains FEPs from seven programs and is the most complete attempt, internationally, at compiling a comprehensive list of FEPs potentially relevant to radioactive waste disposal. Consistent with the diverse backgrounds of the waste disposal programs contributing to the NEA list, FEPs were identified by a variety of methods, including expert judgment, informal elicitation, event tree analysis, stakeholder review, and regulatory stipulation.

Version 1.0 of the NEA database exists in draft form only. It contains extensive descriptions of potentially relevant FEPs from each of the seven programs, along with program-specific technical discussions regarding their applicability. The YMP FEPs list includes the relevant portions of each of the NEA FEPs but does not include the program-specific details unless they are also relevant to YMP. SAM (n.d. [139333], Section B-2.3) identifies the publications listed in Table B-2 as the basis for the NEA FEPs. However, in many cases the draft NEA database contains more extensive FEP descriptions than the supporting publications. The number of FEPs in the database from each of these international programs is also listed in Table B-2.

Table B-2. Origin of the 1,261 Features, Events, and Processes in the U.S. Nuclear Energy Agency Database

Nation	Organization	Type of Study	Number of FEPs[a]	Reference
Canada	Atomic Energy of Canada, Ltd. (AECL)	Scenario Analysis	281	Goodwin et al. 1994 [100983]
International	U.S. Nuclear Energy Agency (NEA)	Scenario Working Group	146	Nuclear Energy Agency 1992 [100479]
Sweden	Swedish Nuclear Power Inspectorate (SKI)	SITE-94	106	Chapman et al. 1995 [100970]
Sweden	Joint SKI and Swedish Nuclear Fuel and Waste Management Co. Svensk Karnbranslehantering AB (SKB)	Scenario Development	158	Andersson 1989 [100956]
United Kingdom	Her Majesty Inspectorate of Pollution (HMIP)	Intermediate- and low-level waste disposal	79	Miller and Chapman 1993 [100996]
Switzerland	National Cooperative for the Disposal of Radioactive Waste (NAGRA)	Kristallin-1	245	NAGRA 1994 [124260]
United States	Waste Isolation Pilot Plant (WIPP)	Compliance Application	246	DOE 1996 [100975]

NOTE: [a] These include FEPs from both the cited reference and the draft NEA database.

B.2.2 YMP-SPECIFIC FEPS

The 1,261 NEA FEPs in the YMP FEP list were supplemented with 292 YMP-specific FEPs identified in a search of YMP literature (Barr 1999 [139292]). Because the YMP is the only potential repository proposed for an unsaturated fractured tuff, many of these FEPs represent events and processes not otherwise included in the international compilation. The 1988 Site Characterization Plan (DOE 1988 [100282], Section 8.3.5.13) itemized 99 specific issues, from which 91 YMP-specific FEPs were identified. The other eight issues were considered to be better captured or subsumed in other similar, but more broadly defined, FEPs. Other project documents provided the general basis for 201 additional YMP-specific FEPs, as described in "Origin of Yucca Mountain FEPs in the database prior to the last set of workshops" (Barr 1999 [139292]). The origin of the 292 YMP-specific FEPs are summarized in Table B-3.

Table B-3. Origin of the 292 Features, Events, and Processes Identified by a Review of the Yucca Mountain Project Literature

Source Document	Number of FEPs	Reference
YMP Site Characterization Plan (YSCP)	91	DOE 1988 [100282]
Other YMP Documents	201	Barr 1999 [139292]

B.2.3 ITERATIVE CIVILIAN RADIOACTIVE WASTE MANAGEMENT SYSTEM MANAGEMENT AND OPERATING CONTRACTOR REVIEW OF THE YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES LIST

The resulting YMP list of 1,553 FEPs identified from the NEA database and YMP literature was taken to a series of technical workshops convened between December 1998 and April 1999 (Table B-4). At these workshops, the FEPs relevant to each subject area were reviewed and discussed by subject matter experts within the project. During these reviews and the associated intensive discussions, workshop participants identified 82 additional YMP-specific FEPs, as summarized in Table B-4. Workshop participants also proposed several issues that were related to FEPs already in the database, in which case, the existing FEP descriptions were expanded to include the new issues.

Table B-4. Origin of the 82 Features, Events, and Processes Identified at the Yucca Mountain Project Workshops Held between December 1998 and April 1999

Workshop	Date	Number of FEPs	Reference
Unsaturated-Zone Flow and Transport (UZ)	Dec. 14-16, 1998	0	[b]
DOE Spent Nuclear Fuel (DSNF) FEPs	Jan. 19, 1999	40	Eide 2000 [149435]
Waste Form (WF)	Feb. 2-4, 1999	12	[a]
Disruptive Events (DE)	Feb. 9-11, 1999	18 6	CRWMS M&O1998 [101095][a]
Saturated Zone Flow/Transport and Biosphere (SZ/Bio)	Feb. 17-19, 1999	1	[a]

Table B-4. Origin of the 82 Features, Events, and Processes Identified at the Yucca Mountain Project Workshops Held between December 1998 and April 1999 (Continued)

Workshop	Date	Number of FEPs	Reference
Thermal-Hydrology (TH) and Coupled Processes	Mar. 24-25, 1999	1	a
In-Drift Geochemical Environment (IDGE) and EBS Transport	Apr. 13-15, 1999	2	a
Waste Package Degradation (WP)	Apr. 20-21, 1999	2	a

NOTES: [a] Indicates that new FEPs were generated by roundtable discussions and subsequently entered directly into the database.

[b] Indicates that no new FEPs were generated at this workshop.

Except for the 40 FEPs from the DSNF Workshop and 18 criticality-related FEPs from the DE Workshop, these additional YMP-specific FEPs were developed informally during roundtable discussions at the workshops and have no formal documentation. Eide (2000 [149435], Tables 1 and 2) documents 25 YMP DSNF-related FEPs derived using a master logic diagram (MLD) approach and an additional 15 DSNF FEPs derived using a comparison approach (CA) between DSNF and commercial spent nuclear fuel (CSNF). The origin of the 18 criticality FEPs from The *Disposal Criticality Analysis Methodology Topical Report* (CRWMS M&O 1998 [101095], Section 3.1) is noted in specific entries in the database. These FEPs include in-situ criticality (ISC), near-field criticality (NFC), and far-field criticality (FFC).

A second round of reviews by subject matter experts was performed in 1999 and 2000, in association with the development of FEP AMRs (listed in Table B-1). During the preparation of the FEP AMRs, subject matter experts reviewed the existing FEPs relevant to their subject area and, where necessary, identified new or missing FEPs. This review and documentation process identified nine additional FEPs, as summarized in Table B-5.

Table B-5. Origin of the Nine Features, Events, and Processes Identified in Features, Events, and Process Analysis Model Reports

FEP AMR Subject Area and ID		Number of FEPs	Reference
WF Misc.	ANL-WIS-MD-000009	2	CRWMS M&O 2000 [146498]
WF Clad	ANL-WIS-MD-000008	2	CRWMS M&O 2000 [150099]
WF Col	ANL-WIS-MD-000012	3	CRWMS M&O 2000 [125156]
EBS	ANL-WIS-PA-000002	2	CRWMS M&O 2000 [136951]

For FEPs related to EBS degradation, flow, and transport, a systematic top-down study (CRWMS M&O 2000 [146680]) was performed to identify any potential FEPs not on the list of FEPs distributed to the EBS FEP AMR (CRWMS M&O 2000 [136951]). The results of the top-down study confirmed the existing EBS-related FEPs and identified the two new FEPs noted in Table B-5, which were incorporated into the EBS FEP AMR (CRWMS M&O 2000 [136951]).

B.2.4 EXTERNAL REVIEW OF THE YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES LIST

An interim version of the YMP FEP list was provided to the NRC in association with the NRC/DOE Appendix 7 Meeting on the FEPs Database held September 8, 1999. A subsequent NRC audit of this interim version of the YMP FEP list identified one potential FEP unrelated to any existing FEPs (Pickett and Leslie 1999 [150373], Section 3.3). The audit also identified three potential FEPs that were possibly related to existing FEPs. Two of these FEPs were subsequently determined to be redundant to or subsumed in existing FEPs. The other two FEPs, noted in Table B-6, were added to the YMP FEP list.

Table B-6. Origin of the Two Features, Events, and Processes Identified in External Reviews

Review	Number of FEPs	Reference
NRC NFE Audit	2	Pickett and Leslie 1999 [150373]

B.2.5 FUTURE DEVELOPMENT OF THE YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES LIST

While the FEPs catalogued in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D) are considered to be reasonably comprehensive, the YMP FEP list is open and may continue to expand if additional FEPs are identified either, within the CRWMS M&O and DOE or from external sources. New FEPs, if identified, will be incorporated into subsequent revisions of the database.

B.3. YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES CLASSIFICATIONS

B.3.1 DATABASE STRUCTURE

Many FEP classification schemes are possible, and there is no inherently correct way to order FEPs. The structure of the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]) follows the NEA classification scheme (SAM n.d. [139333], Section 3), in which FEPs are organized under a hierarchical structure of layers, categories, and headings. The NEA structure comprises a comprehensive group of subject areas potentially relevant to radioactive waste disposal that was developed to systematically classify the FEPs from seven different international programs (Section B-2.1). The NEA classification scheme was selected because it maintains consistency between NEA and YMP databases, which facilitates reviewing for completeness.

The structure of the NEA FEP Database Version 1.0 is defined by 4 layers, 12 categories, and 134 headings. The search of YMP literature for FEPs by Barr (1999 [139292]) identified an additional heading relevant to YMP (the Nuclear Criticality heading in the Geologic Environment category) that was not in the NEA database. Therefore, the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]) has 4 layers, 12 categories, and 135 headings. The hierarchical relationship between these layers, categories, and headings is shown in Table B-7.

Table B-7. Hierarchical Structure of the Yucca Mountain Project Features, Events, and Processes Database

Layers	Categories	Headings [a]
0. Assessment Basis		0.1.01 Impacts of concern 0.1.02 Time scales 0.1.03 Spatial domain 0.1.04 Potential repository assumptions 0.1.05 Future human action assumptions 0.1.06 Future human behavior assumptions 0.1.07 Dose response assumptions 0.1.08 Aims of the assessment 0.1.09 Regulatory requirements and exclusions 0.1.10 Model and data issues
1. External Factors	1.1 Repository Issues	1.1.01 Site investigation 1.1.02 Excavation/construction 1.1.03 Emplacement of wastes 1.1.04 Closure and sealing 1.1.05 Records and markers 1.1.06 Waste allocation 1.1.07 Design 1.1.08 Quality control 1.1.09 Schedule and planning 1.1.10 Administrative control of site 1.1.11 Monitoring 1.1.12 Accidents and unplanned events 1.1.13 Retrievability
	1.2 Geologic Processes and Effects	1.2.01 Tectonic movements 1.2.02 Deformation 1.2.03 Seismicity 1.2.04 Volcanic activity 1.2.05 Metamorphism 1.2.06 Hydrothermal activity 1.2.07 Erosion and sedimentation 1.2.08 Diagenesis 1.2.09 Salt diapirism and dissolution 1.2.10 Hydrologic response to geologic changes
	1.3 Climatic Processes and Effects	1.3.01 Climate change, global 1.3.02 Climate change, regional 1.3.03 Sea level changes 1.3.04 Periglacial effects 1.3.05 Glacial and ice sheet effects 1.3.06 Warm climate effects 1.3.07 Hydrologic response to climate change 1.3.08 Ecological response to climate change 1.3.09 Human response to climate change

Table B-7. Hierarchical Structure of the Yucca Mountain Project Features, Events, and Processes Database (Continued)

Layers	Categories	Headings [a]
1. External Factors	1.4 Future Human Actions (Active)	1.4.01 Human influences on climate 1.4.02 Inadvertent/deliberate human actions 1.4.03 Unintrusive site investigation 1.4.04 Drilling activities 1.4.05 Mining and other underground activities 1.4.06 Surface environment 1.4.07 Water management (wells, reservoirs) 1.4.08 Social developments 1.4.09 Technological developments 1.4.10 Remedial actions 1.4.11 Explosions and crashes
	1.5 Other	1.5.01 Meteorite impact 1.5.02 Species evolution 1.5.03 Miscellaneous (earth tides)
2. Disposal System Domain: Environmental Factors	2.1 Wastes and Engineered Features	2.1.01 Inventory 2.1.02 Waste form 2.1.03 Waste container 2.1.04 Backfill 2.1.05 Seals, cavern/tunnel/shaft 2.1.06 Other features (drip shield, invert) 2.1.07 Mechanical processes and conditions 2.1.08 Hydrogeologic processes and conditions 2.1.09 Geochemical processes and conditions 2.1.10 Biological processes and conditions 2.1.11 Thermal processes and conditions 2.1.12 Gas sources and effects 2.1.13 Radiation effects 2.1.14 Nuclear criticality
	2.2 Geologic Environment	2.2.01 Excavation disturbed zone 2.2.02 Host rock 2.2.03 Geologic units, other 2.2.04 Discontinuities, large scale 2.2.05 Contaminant transport pathways 2.2.06 Mechanical processes and conditions 2.2.07 Hydrogeologic processes and conditions 2.2.08 Geochemical processes and conditions 2.2.09 Biological processes and conditions 2.2.10 Thermal processes and conditions 2.2.11 Gas sources and effects 2.2.12 Undetected features 2.2.13 Geological resources 2.2.14 Nuclear criticality

Table B-7. Hierarchical Structure of the Yucca Mountain Project Features, Events, and Processes Database (Continued)

Layers	Categories	Headings [a]
2. Disposal System Domain: Environmental Factors	2.3 Surface Environment	2.3.01 Topography 2.3.02 Soil 2.3.03 Aquifers/water-bearing features, near surface 2.3.04 Lakes, rivers, streams, springs 2.3.05 Coastal features 2.3.06 Marine features 2.3.07 Atmosphere 2.3.08 Vegetation 2.3.09 Animal populations 2.3.10 Meteorology 2.3.11 Hydrologic regime and water balance 2.3.12 Erosion and deposition 2.3.13 Ecological/biological/microbial systems
	2.4 Human Behavior	2.4.01 Human characteristics 2.4.02 Adults, children, infants 2.4.03 Diet and fluid intake 2.4.04 Habits, nondiet-related 2.4.05 Community characteristics 2.4.06 Food and water processing and preparation 2.4.07 Dwellings 2.4.08 Wild/natural land and water use 2.4.09 Rural/agricultural land and water use 2.4.10 Urban/industrial land and water use 2.4.11 Leisure and other uses of the environment
3. Disposal System Domain: Radionuclide/ Contaminant Factors	3.1 Contaminant Characteristics	3.1.01 Radioactive decay and ingrowth 3.1.02 Chemical/organic toxin stability 3.1.03 Inorganics 3.1.04 Volatiles 3.1.05 Organics 3.1.06 Noble Gases
	3.2 Contaminant Release/ Migration Factors	3.2.01 Dissolution, precipitation, crystallization 3.2.02 Speciation and solubility 3.2.03 Sorption/desorption processes 3.2.04 Colloids 3.2.05 Chemical/complexing agents, effect on transport 3.2.06 Microbiological/plant-mediated processes 3.2.07 Water-mediated transport 3.2.08 Solid-mediated transport 3.2.09 Gas-mediated transport 3.2.10 Atmospheric transport 3.2.11 Animal, plant, microbe mediated transport 3.2.12 Human-action-mediated transport 3.2.13 Food chains, uptake of contaminants

Table B-7. Hierarchical Structure of the Yucca Mountain Project Features, Events, and Processes Database (Continued)

Layers	Categories	Headings [a]
3. Disposal System Domain: Radionuclide/ Contaminant Factors	3.3 Exposure Factors	3.3.01 Drinking water, food, drugs, concentrations 3.3.02 Environmental media, concentrations 3.3.03 Nonfood products, concentrations 3.3.04 Exposure modes 3.3.05 Dosimetry 3.3.06 Radiological toxicity/effects 3.3.07 Nonradiological toxicity/effects 3.3.08 Radon exposure

Source: CRWMS M&O 2000 [150806]

NOTE: [a] Some heading descriptions are paraphrased.

Each of the 1,646 FEPs in the YMP FEP list identified in Section B.2 of this Appendix was assigned (mapped) to a single heading in the YMP FEP Database. For the 1,261 FEPs adopted from other international programs (Table B-2), preliminary mappings were based on the relationships identified in the NEA database, although some adjustments were made to reflect YMP-specific conditions. The task of finding unique mappings was complicated by the fact that many FEPs in the NEA database are mapped to multiple headings. In cases where more than one heading was identified, the most relevant one for YMP was selected, and cross-references were made to the others. This approach eliminated duplicative entries in the YMP FEP Database. For the 385 YMP-specific FEPs (Tables B-3 through B-6), which are not included in the NEA database, preliminary mappings were made to the most relevant heading. The preliminary mappings were reviewed during the December 1998 to April 1999 workshops (Table B-4) and during preparation of the FEP AMRs (Table B-1), and some changes in mapping were made as defined by subject matter experts.

Each of the 1,646 FEPs in the YMP FEP list is an individual entry (record) in the YMP FEP Database, as are the 151 layer, category, and heading entries that define the YMP FEP classifications. Therefore, the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D) contains a total of 1,797 individual entries. The mapping of FEP entries to the heading entries resulted in a database where all related entries were grouped together under the same classification heading (with overarching categories and levels). Links between database entries and specific FEP AMR PMR subject areas (see Section B.3.4) allow for additional groupings to be examined. A further categorization of the entries, to better facilitate systematic screening, is described in Section B.3.2.

B.3.2 PRIMARY AND SECONDARY FEATURES, EVENTS, AND PROCESSES

There is no uniquely correct level of detail at which to define and (or) aggregate FEPs. In the case where FEPs are too narrowly defined, it is infeasible to develop specific screening decisions for each FEP. Instead, it becomes more efficient to develop more broadly based screening decisions that apply to multiple, related FEPs. In cases where FEPs are too coarsely defined, it becomes difficult to isolate important subissues, and, consequently, some important subissues may get excluded, while other unimportant issues may get included. For efficiency, FEPs need to be aggregated at the coarsest level at which technically sound screening decisions can be made, while still maintaining adequate detail for the purposes of the analysis.

The all-inclusive bottom-up approach used to develop the YMP FEP list resulted in considerable redundancy in the FEP list, because the same FEPs were frequently identified by multiple sources. This was especially true of the international FEPs, where each of the seven programs would often identify the same FEP (e.g., meteorite impact). It was also true of the YMP-specific FEPs (and some of the more general international FEPs), where variations of the same FEP would be identified in various literature or reviews.

To eliminate the redundancy and to create a more efficient aggregation of FEPs to carry forward into the screening process (Section B.4), each of the 1,797 entries catalogued in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D) was further identified as either a primary, secondary, or classification (layer, category, or heading) entry. Assignments to each of the three types of entries were based on the follow criteria:

Primary FEP Entry–These are database entries that encompass a single process or event, or a few closely related or coupled processes or events, that can be addressed by a specific screening discussion. Each primary FEP is addressed by a YMP-specific screening discussion taken from one or more FEP AMRs. A primary FEP may also include one or more related secondary FEPs that are covered by the same screening discussion.

Secondary FEP Entry–These are database entries that are (1) redundant to another FEP (e.g., several NEA contributors identified the same FEP), (2) specific to another program (captured more generally in a different YMP-specific FEP), or (3) better captured or subsumed in another similar, but more broadly-defined, YMP-specific FEP. Each secondary FEP is mapped to a primary FEP and must be completely addressed by the screening discussion of that primary FEP.

Classification (Layer, Category, Heading) Entry–These are database entries that represent the hierarchical levels of classification within the database (see Table B-7). Classification entries are neither primary FEPs nor secondary FEPs. They are defined too broadly to be addressed by a single screening discussion (as with a primary FEP) and cannot be encompassed by an overlying FEP (as with a secondary FEP). Rather, they classify one or more underlying, related, primary FEPs and do not require screening discussions.

Based on the preliminary mapping of the FEP entries to the heading entries (described in Section B.3.1), a preliminary attempt was made to identify primary, secondary, and classification entries. The following steps were followed:

1. The 4 layer, 12 category, and 135 heading entries were initially defined as classification entries (as described in Step 4, below, some heading entries were subsequently re-classified as primary FEPs).

2. The FEP entries mapped under each heading were informally separated into groups of related FEPs (e.g., under 2.1.03 Waste Container were such groupings as corrosion, mechanical damage, and early failures).

3. Each of the informal groupings of related FEPs from step 2, above, was further evaluated to identify FEPs that would likely require separate screening discussions. These independent FEPs were identified as primary FEPs (with no associated secondary FEPs).

4. In some cases, the informal groupings of FEPs under a specific heading entry were closely enough related that they could all be addressed by a screening discussion at the overlying heading level. In these cases, the heading entry (previously defined as a classification entry in step 1, above) was designated as a primary FEP. The underlying FEPs were designated as secondary FEPs to the heading level primary FEP.

5. Each of the remaining informal groupings of related FEPs from step 2, above, (that were not mapped as independent in step 3, above, or heading level in step 4, above) was further evaluated to better identify (a) multiple FEPs covering related or coupled processes or events that could likely be addressed by a single screening discussion, or (b) redundant FEPs. The resulting groups of FEPs were each selected to be represented by a primary FEP.

6. Each of the primary FEP groups identified from step 5, above, was examined to select a specific primary FEP. The primary FEP was chosen from the group of related or redundant FEPs as the FEP that best represented and was most inclusive of the group of FEPs as a whole. The other FEPs in the group were designated as secondary FEPs to the selected primary FEP.

7. For each of the primary FEPs (selected in steps 3, 4, and 6, above), a YMP primary FEP description was prepared. This description was based on the FEP description provided by the originator (e.g., the NEA database or YMP literature). The originator description was (a) edited to ensure that it was specific to YMP, and (b) expanded to ensure that all aspects of the related secondary FEPs were also addressed.

Because any categorization of FEPs is subjective, the preliminary identification of primary, secondary, and classification entries was reviewed by subject matter experts. During the December 1998 to April 1999 workshops (Table B-4), some primary and secondary categorizations were revised, and some of the FEPs were remapped to different headings. During preparation of the FEP AMRs (Table B-1), additional changes to primary and secondary FEP mappings and to the YMP primary FEP descriptions were identified. The FEP AMRs also confirmed that the remaining mappings were appropriate and that the YMP primary FEP descriptions did encompass all aspects of the related secondary FEPs.

After all the reviews and confirmations, the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]) contains 111 classification entries (151, less 40 heading entries that are also primary FEPs), 323 primary FEP entries (including the 40 headings) and 1,363 secondary FEP entries.

The objective of the categorization into primary, secondary, and classification entries was to identify a subset of FEP entries, the primary FEPs, which capture all of the issues relevant to the postclosure performance of the potential Yucca Mountain repository and that can be addressed at an appropriate level of screening. As a result of the categorization described in this section, it was only necessary to develop screening decisions and supporting documentation (as described in Section B.4) for the 323 primary FEPs, not for all 1,797 YMP FEP list entries. A minor exception was found in the input AMRs. Two secondary FEPs—2.1.02.08.04 and 1.4.01.03.01—were addressed explicitly. All other secondary FEPs were screened at the overlying primary FEP level.

B.3.3 ORGANIZATION AND NUMBERING OF DATABASE ENTRIES

The organization of the FEP entries within the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]) to follow the NEA hierarchical structure is controlled by the YMP FEP database number associated with each FEP entry. This number has the form x.x.xx.xx.xx and defines classification (layer, category, heading), primary, and secondary entries as follows:

- x.0.00.00.00 Layer
- x.x.00.00.00 Category
- x.x.xx.00.00 Heading (some of these are also Primary FEPs)
- x.x.xx.xx.00 Primary FEP (where the first number x.x.xx is the overlying Heading)
- x.x.xx.xx.xx Secondary FEP (where the first number x.x.xx.xx is the overlying primary FEP).

With this numbering scheme, the YMP FEP database number always identifies to which heading a primary FEP is mapped and to which primary FEP a secondary FEP is associated.

B.3.4 DATABASE FIELDS

For each of the 1,797 entries in REV00 of the database, there are 26 data/text fields. Each of these fields is described below. Fields that contain input or confirmation from the FEP AMRs are noted with a double underline.

YMP FEP Database Number–This is a numeric identifier that places the FEP in the proper location within the database structure. The numbering scheme follows a hierarchical structure classifying FEPs into layers (x...), categories (x.x...), headings (x.x.xx...), primary FEPs (x.x.xx.xx...), and secondary FEPs (x.x.xx.xx.xx).

FEP Name–This is a short, descriptive title of an FEP.

FEP Class–This is the identification used for primary, secondary, and classification (layer, category, heading) entries. Primary FEPs are those FEPs for which the YMP has developed and documented screening discussions. Secondary FEPs are mapped to primary FEPs, either because they are redundant with the associated primary FEP, or because they represent a subcase of the primary FEP that is more effectively addressed at a higher level. Secondary FEPs are retained in the database for completeness, but users of the database are referred to the related Primary FEPs for the screening discussions.

Related FEPs–This is the identification used for entries containing related information. For primary FEPs, other related primary FEPs (if any) are listed. For secondary FEPs, the associated primary FEP is listed. However, for layer, category, and heading classification entries, underlying headings are assumed to be related and are not listed explicitly.

Source Identifier–This is the alphanumeric identifier that provides traceability to the originator (e.g., NEA contributing program, YMP workshop, FEP AMR) as shown in Table B-8. Note that the Source Identifier is not related to the NEA structure or YMP FEP Database Number.

Table B-8. Abbreviations Used in Source Identifier Field

Source (see Tables B-2 through B-6)	Source Identifier Format
AECL	Ax.xxx
NEA	Nx.x.xx
SKI/SKB	Jx.x.xx
SKI	Sxxx
HMIP	HMIPx.x.x
NAGRA	Kx.xx
DOE-WIPP	Wx.xxx
YMP Site Characterization Plan (YSCP)	YSCPxx
Other YMP Documents	Ymxx
UZ Workshop	UZ/xxxx
DSNF Workshop	CA-x, MLD-x
WF Workshop	WF/xxxx
DE Workshop	DE/xxxx, ISC-x, NFC-x, FFC-x
SZ/Bio Workshop	SZ/xxxx, BIO/xxxx
TH Workshop	TH/xxxx
IDGE Workshop	ID/xxxx
WP Workshop	WP/xxxx
NEA Layer, Category, Heading	NEA xxxxxxxx
Other Layer, Category, Heading	Non-NEA xxxxxxxx
WF Miscellaneous FEP AMR	WF Misc AMR-x
WF Cladding FEP AMR	WF Clad AMR-x
WF Colloid FEP AMR	WF Col AMR-x
EBS FEP AMR	EBS AMR-x
NRC NFE Audit	NRC-x

Source: Table B-1

NEA Category–This is the alphanumeric used for identifying the preliminary mapping of the FEPs relative to the NEA database headings. This field is based on preliminary mapping and has been superceded by the YMP FEP Database Number field. It is retained only for traceability to earlier versions of the database. Note that for new FEPs that were identified during and subsequent to the December 1998 to April 1999 workshops, the Source Identifier is repeated in the NEA Category field.

YMP Primary FEP Description–This is the description of each FEP and its potential relevance to YMP, typically edited from the originator description. Where secondary FEPs are associated with a primary FEP, the description also includes all of the FEPs described by the secondary FEPs.

Originator FEP Description–This is the verbatim text of an FEP description from originator documentation. The originator is noted in parentheses, where possible.

Screening Decision–This is a statement of whether the FEP is included in the quantitative Total System Performance Assessment (TSPA) models or excluded from the TSPA on specific criteria provided by the regulations.

Screening Argument–This is a summary discussion of the technical basis for the Screening Decision, with citations to appropriate AMRs. (For excluded FEPs, this is the key text.)

TSPA Disposition–This is a summary discussion of the treatment of the FEP in the TSPA, with citations and cross-references to the appropriate AMRs. (For included FEPs, this is the key text.)

PMR–This identifies the PMR subject area that was assigned initial responsibility for technical evaluation of the FEP. This field was not updated for REV00. Instead, the subject area where the FEP was ultimately addressed is listed in the Input AMR field.

Input AMR–This identifies the FEP AMR where the qualified screening discussion is documented. Verbatim text for several fields, including the Screening Decision, Screening Argument, TSPA Disposition, Supplemental Discussion, and References, are taken from the Input AMR. The Input AMR identifier also indicates the subject area in which the FEP is grouped.

IRSR–This identifies NRC IRSR subissues related to the FEP.

Supplemental Discussion–This discussion provides additional information supporting the Screening Decision beyond what is summarized in the Screening Argument and TSPA Disposition fields.

References–These identify the references cited in the Screening Argument and (or) TSPA Disposition summaries.

Modified by–The name of last person to modify an FEP record appears here.

Mod Date–The date of the last modification to an FEP record appears here.

Mod Time–The time of the last modification to an FEP record appears here.

Record Number–This is the numeric identifier of the record sequence.

F Keyword–This is an identifier feature keyword from a specified list that is used for keyword searches. For REV 00, this field is blank.

E Keyword–This is an identifier event keyword from a specified list that is used for keyword searches. For REV 00, this field is blank.

P Keyword–This is an identifier process keyword from a specified list that is used for keyword searches. For REV 00, this field is blank.

Workshop–This identifies all of the Workshops held between December 1998 to April 1999, where the FEP was reviewed and discussed. This field is retained only for traceability back to preliminary versions of the database.

Owner–This is the name of the technical, subject-matter expert given responsibility to address the FEP at the December 1998 to April 1999 workshops. This field has been superceded by the Input AMR field, which now establishes FEP ownership. For REV 00, this field is blank.

Notes–These consist of miscellaneous notes and comments related to the FEP.

B.4. YUCCA MOUNTAIN PROJECT AND FEATURES, EVENTS, AND PROCESSES SCREENING CRITERIA AND GUIDELINES

B.4.1 SCREENING CRITERIA

Each primary FEP (and, by association, each secondary FEP) was screened for inclusion or exclusion in the TSPA on the basis of three criteria, developed from DOE's Interim Guidance (Dyer 1999 [105655]). The three criteria are as follows:

1. **Regulatory**–DOE's Interim Guidance (Dyer 1999 [105655], Subpart E) provides regulatory guidance regarding certain assumptions about the TSPA. Some FEPs may be specifically exempted from consideration in TSPA because they are not in accordance with this regulatory guidance or are not applicable by regulation. FEPs that are inconsistent with the regulatory assumptions may be excluded (screened out) from the TSPA by regulation. The most notable examples are the regulatory specification of the human intrusion scenario and the critical group characteristics. Any FEPs which invoke human intrusion scenarios or critical group characteristics that are inconsistent with what is specified in the regulations are screened out by regulation.

2. **Probability**–The probability criterion is stated in DOE's Interim Guidance (Dyer 1999 [105655], Section 114):

    a. Consider only events that have at least one chance in 10,000 of occurring over 10,000 years.

    b. FEPs with a lower probability of occurrence may be excluded (screened out) from the TSPA on the basis of low probability.

3. **Consequence**–The consequence criteria are stated in DOE's Interim Guidance (Dyer 1999 [105655], Section 114):

   a. Provide the technical basis for either inclusion or exclusion of specific FEPs of the geologic setting in the performance assessment. Specific FEPs of the geologic setting must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

   b. Provide the technical basis for either inclusion or exclusion of degradation, deterioration, or alteration processes of engineered barriers in the performance assessment, including those processes that would adversely affect the performance of natural barriers. Degradation, deterioration, or alteration processes of engineered barriers must be evaluated in detail if the magnitude and time of the resulting expected annual dose would be significantly changed by their omission.

FEPs whose exclusion would not significantly change the expected annual dose may be excluded (screened out) from the TSPA on the basis of low consequence.

B.4.2 SCREENING GUIDELINES AND IMPLEMENTATION

Because DOE's Interim Guidance (Dyer 1999 [105655], Section 114) allows exclusion of FEPs on the basis of either low probability or low consequence, an FEP need not be shown to be both of low probability and low consequence to be excluded. Therefore, the order in which the criteria are applied is not essential. In some cases, a component of the FEP was included while another component of the FEP was excluded. In practice, regulatory criteria are examined first, and then, at the discretion of the analyst, either probability or consequence criteria are examined next.

As noted in Section B-1, the FEP screening was performed by subject matter experts and documented in FEP AMRs (listed in Table B-1). Specific screening data from the FEP AMRs was then imported into the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D), in accordance with the data transfer controls (CRWMS M&O 1999 [150395]). The screening data are catalogued in the database. The verification of the technical accuracy and completeness of the screening data is the responsibility of the FEP AMRs.

The specific database fields containing screening data from the FEP AMRs were identified in Section B-3.4. To satisfy the screening criteria of DOE's Interim Guidance (Dyer 1999 [105655], Section 114) and to satisfy the TSPAI IRSR subissues pertaining to FEPs and scenario analysis (NRC 2000 [149372], Sections 4.1.1.2 and 4.2), guidelines have been established for the content of four of these fields: YMP Primary FEP Description, Screening Decision, Screening Argument, and TSPA Disposition. Because the technical defensibility of the content of these fields is the responsibility of the FEP AMRs, the content cannot be changed outside of the FEP AMRs. Therefore, these guidelines apply to the FEP AMRs. Key aspects of the guidelines are summarized below:

YMP Primary FEP Description–It must be relevant to YMP and must include all of the related FEPs identified in associated secondary FEPs.

Screening Decision–It must state whether the FEP is included or excluded from the TSPA.

For excluded FEPs, the exclusion criteria (regulation, low probability, low consequence) must be explicitly identified.

For partially included and (or) partially excluded FEPs, the various components that are included and excluded must be identified (e.g., FEP 1.2.02.01.000, Fractures, includes the effects of the present-day fracture system but excludes the effects of changes to the fracture system on the basis of low consequence).

Screening Argument–For excluded FEPs, this is the main screening discussion. A summary of the technical basis for exclusion must be presented, and the summary must address all secondary FEP issues.

Low probability exclusions must include an explicit comparison of the probability of occurrence to the regulatory criteria ($<10^{-4}$ in 10,000 years). The probability must be quantified where possible, although nonquantitative, low-probability arguments are acceptable for not credible FEPs.

Low consequence exclusions must include an explicit statement that there is "no significant change in the expected annual dose." The change in expected annual dose must be quantified where possible, and the interpretation of significant change must be described. (It may be different for each FEP.) It is acceptable to quantify the change in an intermediate performance measure (e.g., radionuclide mass release to the SZ). However, in that case, the qualitative link to change in expected annual dose must be explicitly stated.

Regulatory exclusions must identify a specific regulation and clearly state the rationale for the exclusion.

TSPA Disposition–For included FEPs, this is the main screening discussion. A summary discussion of the treatment of the FEP in the TSPA must be presented. A statement of the scenario class, model and(or) abstraction is desirable.

In some cases, a primary FEP may affect multiple facets of the project and may be relevant to more than one FEP AMR subject area, or it may not fit neatly within the FEP AMR structure. In these cases, rather than create multiple, separate FEPs, the FEP was assigned to more than one FEP AMR. These shared FEPs then had separate screening discussions prepared in the separate FEP AMRs. While informal meetings were held to resolve any contradictory screening discussions for shared FEPs, the multiple screening discussions input to the database were not integrated. As a result, shared FEPs in REV00 may contain duplicative screening information. Similarly, some FEP AMRs modified the YMP Primary FEP Descriptions to ensure that all implications of the secondary FEPs were subsumed in the YMP Primary FEP Descriptions. Where these modified FEPs were shared FEPs, multiple YMP Primary FEP Descriptions were input to the database but not integrated.

B.4.3 FUTURE DEVELOPMENT OF THE YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES SCREENING DATA

REV01 of the database is planned to be completed to support TSPA-SR REV01, conditional on the completion of REV01 of FEP AMRs where necessary. The FEP screening data in the database may be updated through the following activities:

- **Addition of screening data for criticality FEPs**–As noted in Table B-1, criticality FEP screening data were not available. One or more criticality FEP AMRs are planned. Upon completion, the screening data will be transferred to the database from these FEP AMRs.

- **Addition of screening data for the NRC NFE audit FEPs (Table B-6)**–These FEPs were not assigned to an FEP AMR, and, therefore, REV00 of the database does not contain any screening information for them. The NRC NFE audit FEPs will be assigned to an FEP AMR for inclusion in REV01 of the database.

- **Addition of screening data for FEP 2.2.01.04.00**–Ownership of this FEP was transferred from one FEP AMR to another, but the screening discussion was inadvertently omitted from both affected FEP AMRs. It will be reassigned to the appropriate FEP AMR for inclusion in REV01 of the database.

- **Addition of screening decisions based on the current no-backfill design**–The screening discussions in REV00 of the database are based on a potential repository design that includes backfill. The FEP AMRs will be revised to ICN 1 to add screening discussions for the no-backfill design. This information will be transferred to the database.

- **Integration of screening information and YMP primary descriptions for shared FEPs** (see Section B.4.2).

- **Identification of the scenario class (Nominal, Disruptive, or Human Intrusion) in the Screening Decision field for included FEPs.**

- **Creation of a master list of subject areas where regulatory exclusions may apply, tied to specific regulations**–This master list will enhance the defensibility of regulatory exclusions.

- **Revisions to screening discussions that did not meet the content guidelines outlined in Section B.4.2**–These revisions must be made in the FEP AMRs rather than in the database directly. Reviews of REV00 screening discussions may identify those FEPs requiring revision.

B.5. YUCCA MOUNTAIN PROJECT FEATURES, EVENTS, AND PROCESSES SCREENING DECISIONS AND BASES

The YMP FEP Database REV00 was developed as described in this Appendix and in *The Development of Information Catalogued in REV00 of the YMP FEP Database* (CRWMS M&O 2000 [150806]) and supersedes all prior versions. The FEP AMR subject matter experts reviewed each of their assigned primary FEP entries and the associated secondary FEP entries and produced a screening decision and supporting documentation within their FEP AMRs. The subject matter experts also reviewed and either confirmed, or suggested changes to, the YMP Primary FEP Descriptions, the primary/secondary mappings, and the FEP AMR assignments. The FEP AMRs were used as input to the YMP FEP Database REV00 (CRWMS M&O 2000 [150806]). A subset of this database and the screening decisions and bases, are summarized in Tables B-9 through B-17 for all primary FEPs. The tables are organized by subject area. As noted in Section B.4.2, many FEPs are shared among different subject areas and, therefore, appear in multiple tables. Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses, then the entire FEP is included or excluded, as indicated. Abbreviations used to indicate FEP AMR reports are given in Table B-1.

Table B-9. Screening Decisions and Bases for System Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
0.1.02.00.00	Time scales of concern	SYS Include
0.1.03.00.00	Spatial domain of concern	SYS Include
0.1.09.00.00	Regulatory requirements and exclusions	SYS Exclude – By regulation (Secondary FEPs) Include (Primary FEP)
0.1.10.00.00	Model and data issues	SYS Exclude – Low consequence (unmodeled design features) Include (everything except unmodeled design features)
1.1.05.00.00	Records and markers, repository	SYS Include (initial construction of markers and archiving of records, and for subsequent loss of records) Exclude – Low consequence and by regulation (efficacy of markers and record retention to prevent intrusion after 100-years post-closure)
1.1.07.00.00	Repository design	SYS Include (licensed design and design modifications) Exclude – Low consequence and by regulation (remaining Secondary FEPs) EBS Include (exclude deviations from design)

Table B-9. Screening Decisions and Bases for System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.1.08.00.00	Quality control	SYS Include "Quality Control" (primary FEP and secondary FEPs 1.1.08.00.05 and .06) Exclude – Low consequence / by regulation (remaining Secondary FEPs) EBS Include (exclude defects and deviations)
1.1.09.00.00	Schedule and planning	SYS Exclude – By regulation
1.1.10.00.00	Administrative control, potential repository site	SYS Include (for "Administrative Control, Potential Repository Site" during preclosure period, for initial construction of markers and archiving of records, and for subsequent loss of administrative control) Exclude – By regulation (efficacy of administrative controls beyond 100-years of the postclosure period)
1.1.11.00.00	Monitoring of potential repository	UZ Exclude – Low consequence. SYS Exclude – Low consequence (for monitoring operations) Include (monitoring wells and boreholes are addressed by the human-intrusion scenario)
1.1.12.01.00	Accidents and unplanned events during operation	SYS Exclude – Low consequence EBS Exclude – By regulation
1.1.13.00.00	Retrievability	SYS Include (design elements related to retrievability and emplacement) Exclude (operational and administrative considerations) EBS Include
1.2.05.00.00	Metamorphism	SYS Exclude – Low consequence
1.2.08.00.00	Diagenesis	SYS Exclude – Low consequence
1.2.09.00.00	Salt diapirism and dissolution	SYS Exclude – By regulation, Low consequence
1.2.09.01.00	Diapirism	SYS Exclude – By regulation
1.4.02.01.00	Deliberate human intrusion	SYS Exclude – By regulation (deliberate intrusion) Include (human- intrusion scenario)
1.4.02.02.00	Inadvertent human intrusion	SYS Include – By regulation
1.4.03.00.00	Un-intrusive site investigation	SYS Exclude – Low consequence

Table B-9. Screening Decisions and Bases for System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.4.04.00.00	Drilling activities (human intrusion)	SYS Include (stylized-drilling scenario) Exclude – By regulation (specific-types of drilling scenarios as presented in the secondary FEPs)
1.4.04.01.00	Effects of drilling intrusion	SYS Include (interactions and changes in condition) Exclude – By regulation (materials brought to the surface)
1.4.05.00.00	Mining and other underground activities (human intrusion)	SYS Exclude – By regulation
1.4.11.00.00	Explosions and crashes (human activities)	SYS Exclude – By regulation
1.5.01.01.00	Meteorite impact	SYS Exclude – Low probability (direct exhumation or direct fracturing to repository horizon) Exclude – Low consequence (alteration of flow paths, fracturing of overlying geologic units, and changes in rock stress)
1.5.01.02.00	Extraterrestrial events	SYS Exclude – Low consequence
1.5.03.01.00	Changes in the earth's magnetic field	SYS Exclude – Low consequence
1.5.03.02.00	Earth tides	SYS Exclude – Low consequence
2.2.06.05.00	Salt creep	SYS Exclude – By regulation
2.3.13.03.00	Effects of repository heat on biosphere	SYS Exclude – Low consequence
3.2.10.00.00	Atmospheric transport of contaminants	SYS Include (for various transport mechanisms and species [ashfall]) Exclude – Low consequence (for volatile radionuclides as a gaseous release)
3.3.06.01.00	Toxicity of mined rock	SYS Exclude – By regulation

Source: Table B1

NOTE: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-10. Screening Decisions and Bases for Near-Field Environment Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.1.02.00.00	Excavation / construction	NFE Include (fracture effects) Exclude (chemistry related effects) UZ Include (effects of stress relief and ground support on drift seepage) Exclude – Low consequence (changes in water chemistry) EBS Exclude – Low consequence
1.1.02.02.00	Effects of pre-closure ventilation	NFE Include EBS Include
1.2.02.01.00	Fractures	NFE Include (seepage) Exclude (permanent effects) UZ Include (effects of present-day fracture system) Exclude – Low consequence (effects of changes to the fracture system) SZ Exclude – Low consequence DE Include (existing fracture characteristics) Exclude – Low consequence (changes of fracture characteristics)
2.1.08.01.00	Increased unsaturated water flux at the repository	NFE Include (primary FEP. Climate change is included) Exclude – Low consequence (secondary FEP on water quenching hot waste package) UZ Include EBS Include
2.1.08.02.00	Enhanced influx (Philip's drip)	NFE Include UZ Include EBS Exclude – Low consequence
2.1.08.03.00	Repository dry-out due to waste heat	NFE Include
2.1.08.10.00	Desaturation / dewatering of the repository	NFE Include WF Misc. Include

Table B-10. Screening Decisions and Bases for Near-Field Environment Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.08.11.00	Resaturation of repository	NFE Include EBS Include
2.1.09.01.00	Properties of the potential carrier plume in the waste and EBS	NFE Include WF Misc. Include (potential effects of carrier plume by evaluating the influence of steel corrosion on the water chemistry in order to establish an uncertainty band) Exclude – Low consequence (the changing properties of incoming water, as evaluated by EBS) EBS Include
2.1.09.12.00	Rind (altered zone) formation in waste, EBS, and adjacent rock	NFE Included (in THC model) Excluded – Low consequence (from TH model) WF Misc. Included (in radionuclide mobilization) Excluded (in adjacent rock) EBS Include
2.1.11.01.00	Heat output / temperature in waste and EBS	NFE Include WF Misc. Include EBS Include
2.1.11.02.00	Nonuniform heat distribution / edge effects in repository	NFE Include (Primary FEP) Exclude – Low consequence (TM effects from secondary FEP)
2.2.01.01.00	Excavation and construction-related changes in the adjacent host rock	NFE Exclude – Low consequence UZ Include (the effects of stress relief and ground support on drift seepage) Exclude – Low consequence (changes in water chemistry)
2.2.01.02.00	Thermal and other waste and EBS-related changes in the adjacent host rock	NFE Exclude – Low consequence
2.2.01.03.00	Changes in fluid saturations in the excavation disturbed zone	NFE Exclude – Low consequence
2.2.06.01.00	Changes in stress (due to thermal, seismic, or tectonic effects) change porosity and permeability of rock	NFE Exclude – Low consequence DE Exclude – Low consequence

Table B-10. Screening Decisions and Bases for Near-Field Environment Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.07.10.00	Condensation zone forms around drifts	NFE Include UZ Exclude – Low consequence (mountain-scale effects) Include (effects on drift seepage)
2.2.07.11.00	Return flow from condensation cap / resaturation of dry-out zone	NFE Include (included in process models used in TSPA) UZ Exclude –Low consequence (mountain-scale effects) Include (effects on drift seepage)
2.2.08.03.00	Geochemical interactions in geosphere (dissolution, precipitation, weathering) and effects on radionuclide transport SZ-Groundwater Chemistry FEPs	NFE Include UZ Exclude – Low consequence SZ Include
2.2.08.04.00	Redissolution of precipitates directs more corrosive fluids to containers	NFE Include UZ Include EBS Include
2.2.10.04.00	Thermo-mechanical alteration of fractures near repository	NFE Excluded – Low consequence UZ Exclude – Low consequence
2.2.10.05.00	Thermo-mechanical alteration of rocks above and below the repository	NFE Exclude – Low consequence UZ Exclude – Low consequence
2.2.10.06.00	Thermo-chemical alteration (solubility, speciation, phase changes, precipitation/dissolution) SZ-Groundwater Chemistry FEPs	NFE Exclude – Low consequence (except for the in-drift geochemical model that uses water chemistry and gas-phase composition from the drift-scale THC model that includes thermal-chemical alteration) UZ Exclude – Low consequence SZ Include
2.2.10.10.00	Two-phase buoyant flow / heat pipes	NFE Include UZ Include
2.2.10.12.00	Geosphere dry-out due to waste heat	NFE Include

Table B-10. Screening Decisions and Bases for Near-Field Environment Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.10.13.00	Density-driven groundwater flow (thermal) SZ-Repository Induced Thermal Effects	NFE Include SZ Exclude – Assumed low consequence or probability

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.1.01.01.00	Open site investigation boreholes	UZ Exclude – Low consequence (effects of deep boreholes) Include (effects of ground-support boreholes on drift seepage)
1.1.01.02.00	Loss of integrity of borehole seals	UZ Exclude – Low consequence (effects of deep boreholes) Include (effects of ground-support boreholes on drift seepage)
1.1.02.00.00	Excavation / construction	NFE Include (fracture effects) Exclude (chemistry related effects) UZ Include (effects of stress relief and ground support on drift seepage) Exclude – Low consequence (changes in water chemistry) EBS Exclude – Low consequence
1.1.02.01.00	Site flooding (during construction and operation)	UZ Exclude – Low probability. EBS Exclude – By regulation
1.1.04.01.00	Incomplete closure	UZ Exclude – Low consequence (effects of deep boreholes) Include (effects of ground-support boreholes on drift seepage)
1.1.11.00.00	Monitoring of repository	UZ Exclude – Low consequence. SYS Exclude – Low consequence (for monitoring operations) Include (monitoring wells and boreholes are addressed by the human-intrusion scenario)

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.02.01.00	Fractures	NFE Include (seepage) Exclude (permanent effects) UZ Include (effects of present-day fracture system) Exclude – Low consequence (effects of changes to the fracture system) SZ Exclude – Low consequence DE Include (existing fracture characteristics) Exclude – Low consequence (changes of fracture characteristics)
1.2.02.02.00	Faulting	UZ Include (effects of present-day faults) Exclude – Low consequence (the effects of changes to the faults) SZ Exclude – Low consequence DE Include (existing fault characteristics) Exclude – Low consequence (changes of fault characteristics)
1.2.03.01.00	Seismic activity	UZ Exclude – Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence (indirect effects) Exclude TBV – Low consequence (waste package) Include (drip shield damage and cladding damage)
1.2.04.02.00	Igneous activity causes changes to rock properties	UZ Exclude – Low consequence DE Exclude – Low consequence
1.2.06.00.00	Hydrothermal activity	UZ Exclude – Low probability SZ Exclude – Low consequence
1.2.07.01.00	Erosion / denudation	UZ Exclude – Low consequence Bio Exclude – By regulation
1.2.07.02.00	Deposition	UZ Exclude – Low consequence Bio Exclude – By regulation

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.09.02.00	Large-scale dissolution	UZ Exclude – Low probability SZ Exclude – Low consequence
1.2.10.01.00	Hydrological response to seismic activity	UZ Exclude – Low probability and Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence
1.2.10.02.00	Hydrologic response to igneous activity	UZ Exclude – Low consequence DE Exclude – Low consequence
1.3.01.00.00	Climate change, global	UZ Include Bio Exclude – By regulation
1.3.04.00.00	Periglacial effects	UZ Exclude – Low probability Bio Exclude – By regulation
1.3.05.00.00	Glacial and ice sheet effects, local	UZ Exclude – Low probability Bio Exclude – By regulation
1.3.07.01.00	Drought / water table decline	UZ Exclude – Low consequence SZ Exclude – Low consequence
1.3.07.02.00	Water table rise	UZ Include SZ Include (changes in flux) Exclude – Assumed Low consequence (other effects not included in SZ Flow and Transport)
1.4.01.00.00	Human influences on climate	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.01.01.00	Climate modification increases recharge	UZ Exclude – Low consequence (effects of perched water below repository) Include (effects of increased flux through repository and water table rise)

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.4.01.02.00	Greenhouse gas effects	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.01.03.00	Acid rain	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.01.04.00	Ozone layer failure	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.04.02.00	Abandoned and undetected boreholes	UZ Exclude – Low probability
1.4.06.01.00	Altered soil or surface water chemistry	UZ Exclude – Low probability Bio Exclude – By regulation
2.1.05.01.00	Seal physical properties	UZ Exclude – Low consequence
2.1.05.02.00	Groundwater flow and radionuclide transport in seals	UZ Exclude – Low consequence
2.1.05.03.00	Seal degradation	UZ Exclude – Low consequence
2.1.08.01.00	Increased unsaturated water flux at the repository	NFE Include (primary FEP. Climate change is included) Exclude – Low consequence (secondary FEP on water quenching hot waste package) UZ Include EBS Include
2.1.08.02.00	Enhanced influx (Philip's drip)	NFE Include UZ Include EBS Exclude – Low consequence
2.1.12.01.00	Gas generation	UZ Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.01.01.00	Excavation and construction-related changes in the adjacent host rock	NFE Exclude – Low consequence UZ Include (the effects of stress relief and ground support on drift seepage) Exclude – Low consequence (changes in water chemistry)
2.2.01.05.00	Radionuclide transport in excavation disturbed zone	UZ Exclude – Low consequence
2.2.03.01.00	Stratigraphy	UZ Include SZ Include
2.2.03.02.00	Rock properties of host rock and other units	UZ Include SZ Include
2.2.06.02.00	Changes in stress (due to thermal, seismic, or tectonic effects) produce change in permeability of faults	UZ Exclude – Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence
2.2.06.03.00	Changes in stress (due to seismic or tectonic effects) alter perched water zones	UZ Exclude – Low consequence (effects of perched water changes below the potential repository) Include (effects of perched water changes above potential repository on drift seepage) SZ Exclude – Low consequence DE Exclude – Low consequence
2.2.06.04.00	Effects of subsidence	UZ Exclude – Low consequence
2.2.07.01.00	Locally saturated flow at bedrock/alluvium contact	UZ Include
2.2.07.02.00	Unsaturated groundwater flow in geosphere	UZ Include
2.2.07.03.00	Capillary rise	UZ Include Bio Exclude – By regulation

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.07.04.00	Focusing of unsaturated flow (fingers, weeps)	UZ Include
2.2.07.05.00	Flow and transport in the UZ from episodic infiltration	UZ Exclude – Low consequence (effects of episodic flow resulting from episodic infiltration, effects of transient flow due to thermal-hydrologic processes on radionuclide transport) Include (effects of transient flow due to thermal-hydrologic processes on drift seepage)
2.2.07.06.00	Episodic / pulse release from repository	UZ Exclude – Low consequence (effects of episodic flow resulting from episodic infiltration, effects of transient flow due to thermal-hydrologic processes on radionuclide transport) Include (effects of transient flow due to thermal-hydrologic processes on drift seepage, effects of intermittent waste package failures) EBS Include
2.2.07.07.00	Perched water develops	UZ Exclude – Low consequence (effects of perched water below repository) Include (effects of increased flux through repository, water table rise and present-day perched water)
2.2.07.08.00	Fracture flow in the UZ	UZ Include
2.2.07.09.00	Matrix imbibition in the UZ	UZ Include
2.2.07.10.00	Condensation zone forms around drifts	NFE Include UZ Exclude – Low consequence (mountain-scale effects) Include (effects on drift seepage)
2.2.07.11.00	Return flow from condensation cap / resaturation of dry-out zone	NFE Include (included in process models used in TSPA) UZ Exclude – Low consequence (mountain-scale effects) Include (effects on drift seepage)
2.2.08.01.00	Groundwater chemistry / composition in UZ and SZ SZ-Groundwater Chemistry FEPs	UZ Include (effects of ambient-condition geochemistry) Exclude – Low consequence (changes in geochemical conditions) SZ Include

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.08.02.00	Radionuclide transport occurs in a carrier plume in geosphere SZ-Radionuclide Transport in a Carrier Plume	UZ Exclude – Low consequence SZ Include
2.2.08.03.00	Geochemical interactions in geosphere (dissolution, precipitation, weathering) and effects on radionuclide transport SZ-Groundwater Chemistry FEPs	NFE Include UZ Exclude – Low consequence SZ Include
2.2.08.04.00	Redissolution of precipitates directs more corrosive fluids to containers	NFE Include UZ Include EBS Include
2.2.08.05.00	Osmotic processes	UZ Exclude – Low consequence
2.2.08.06.00	Complexation in geosphere	UZ Include (effects of ambient-condition complexation) Exclude – Low consequence (effects of changes to complex formation due to changes in geochemical conditions) SZ Include
2.2.08.07.00	Radionuclide solubility limits in the geosphere	UZ Exclude – Low consequence SZ Exclude – Low consequence
2.2.08.08.00	Matrix diffusion in geosphere SZ-Matrix Diffusion	UZ Include SZ Include
2.2.08.09.00	Sorption in UZ and SZ	UZ Include SZ Include
2.2.08.10.00	Colloidal transport in geosphere	UZ Include SZ Include

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.09.01.00	Microbial activity in geosphere SZ-Groundwater Chemistry FEPs	UZ Exclude – Low consequence SZ Include
2.2.10.01.00	Repository-induced thermal effects in geosphere	UZ Exclude – Low consequence (mountain-scale thermo-chemical effects) Include (thermo-chemical effects on drift seepage) SZ Exclude – Assumed low consequence or probability
2.2.10.03.00	Natural geothermal effects SZ-Geothermal Effects	UZ Include SZ Include
2.2.10.04.00	Thermo-mechanical alteration of fractures near repository	NFE Excluded – Low consequence UZ Exclude – Low consequence
2.2.10.05.00	Thermo-mechanical alteration of rocks above and below the repository	NFE Exclude – Low consequence UZ Exclude – Low consequence
2.2.10.06.00	Thermo-chemical alteration (solubility, speciation, phase changes, precipitation/dissolution) SZ-Groundwater Chemistry FEPs	NFE Exclude – Low consequence (except for the in-drift geochemical model that uses water chemistry and gas-phase composition from the drift-scale THC model that includes thermal-chemical alteration) UZ Exclude – Low consequence SZ Include
2.2.10.07.00	Thermo-chemical alteration of the Calico Hills unit SZ-Repository Induced Thermal Effects	UZ Exclude – Low consequence SZ Exclude – Assumed low consequence or probability
2.2.10.09.00	Thermo-chemical alteration of the Topopah Spring basal vitrophyre	UZ Exclude – Low consequence
2.2.10.10.00	Two-phase buoyant flow / heat pipes	NFE Include UZ Include

Table B-11. Screening Decisions and Bases for Unsaturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.10.11.00	Natural air flow in the UZ	UZ Exclude – Low consequence
2.2.11.01.00	Naturally-occurring gases in geosphere	UZ Exclude – Low consequence and low probability SZ Exclude – Low consequence
2.2.11.02.00	Gas pressure effects	UZ Exclude – Low consequence and low probability EBS Exclude – Low consequence
2.2.11.03.00	Gas transport in geosphere	UZ Exclude – Low consequence and low probability
2.2.12.00.00	Undetected features (in geosphere) SZ-Undetected features	UZ Exclude – Low consequence and low probability SZ Include
2.3.01.00.00	Topography and morphology	UZ Include
2.3.11.01.00	Precipitation	UZ Include Bio Include (precipitation) Exclude – By regulation (recharge/and climate change)
2.3.11.02.00	Surface runoff and flooding	UZ Include Bio Include (dispersion of contaminants, precipitation, and infiltration) Exclude – By regulation (recharge, water balance)
2.3.11.03.00	Infiltration and recharge (hydrologic and chemical effects)	UZ Exclude – Low consequence (effects of changes to water chemistry) Include (effects of changing infiltration and water table rise)
3.1.01.01.00	Radioactive decay and ingrowth	UZ Include SZ Include WF Misc. Include

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.1.02.00.00	Excavation / construction	NFE Include (fracture effects) Exclude (chemistry related effects) UZ Include (effects of stress relief and ground support on drift seepage) Exclude – Low consequence (changes in water chemistry) EBS Exclude – Low consequence
1.1.02.01.00	Site flooding (during construction and operation)	UZ Exclude – Low probability EBS Exclude – By regulation
1.1.02.02.00	Effects of pre-closure ventilation	NFE Include EBS Include
1.1.02.03.00	Undesirable materials left	EBS Exclude – Low consequence
1.1.03.01.00	Error in waste or backfill emplacement	WP Exclude – Low probability EBS Exclude – By regulation
1.1.07.00.00	Repository design	SYS Include (licensed design and design modifications) Exclude – Low consequence and by regulation (remaining Secondary FEPs) EBS Include (exclude deviations from design)
1.1.08.00.00	Quality control	SYS Include ("Quality Control" primary FEP and secondary FEPs [1.1.08.00.05 and .06]) Exclude – Low consequence / by regulation (remaining Secondary FEPs) EBS Include (exclude defects and deviations)
1.1.12.01.00	Accidents and unplanned events during operation	SYS Exclude – Low consequence EBS Exclude – By regulation
1.1.13.00.00	Retrievability	SYS Include (design elements related to retrievability and emplacement) Exclude (operational and administrative considerations) EBS Include

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.04.03.00	Igneous intrusion into repository	DE Include (as a dike rather than as a sill) EBS Exclude. N/A for EBS
2.1.03.01.00	Corrosion of waste containers	WP Include EBS Include
2.1.03.10.00	Container healing	WP Exclude – Low consequence EBS Include
2.1.03.12.00	Container failure (long-term)	WP Include EBS Include
2.1.04.01.00	Preferential pathways in the backfill	EBS Include
2.1.04.02.00	Physical and chemical properties of backfill	EBS Include
2.1.04.03.00	Erosion or dissolution of backfill	EBS Exclude – Low consequence
2.1.04.04.00	Mechanical effects of backfill	EBS Include
2.1.04.05.00	Backfill evolution	EBS Include
2.1.04.06.00	Properties of bentonite	EBS Exclude – Low (Zero) Probability
2.1.04.07.00	Buffer characteristics	EBS Exclude – Low (Zero) Probability
2.1.04.08.00	Diffusion in backfill	EBS Exclude – Low consequence
2.1.04.09.00	Radionuclide transport through backfill	EBS Exclude – Low consequence
2.1.06.01.00	Degradation of cementitious materials in drift	EBS Include
2.1.06.02.00	Effects of rock reinforcement materials	EBS Include
2.1.06.03.00	Degradation of the liner	EBS Exclude – Low (Zero) Probability
2.1.06.04.00	Flow through the liner	EBS Exclude – Low (Zero) Probability
2.1.06.05.00	Degradation of invert and pedestal	EBS Include

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.06.06.00	Effects and degradation of drip shield	WP Exclude – Low consequence (damage to drip shield by rock fall, damage to drip shield by ground motion during seismic events [TBV pending additional data and/or analysis]) Include (physical and chemical degradation processes) EBS Include
2.1.06.07.00	Effects at material interfaces	WP Include EBS Exclude – Low consequence
2.1.07.01.00	Rockfall (large block) WF Clad-Rockfall	WP Exclude – Low consequence DE Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.07.02.00	Mechanical degradation or collapse of drift	DE Exclude – Low consequence EBS Exclude – Low consequence
2.1.07.03.00	Movement of containers	EBS Include
2.1.07.04.00	Hydrostatic pressure on container	EBS Exclude – Low (Zero) Probability
2.1.07.05.00	Creeping of metallic materials in the EBS	WP Exclude – Low consequence (TBV pending additional inputs and/or analysis from Waste Package Design) EBS Exclude – Low consequence
2.1.07.06.00	Floor buckling	EBS Exclude – Low consequence
2.1.08.01.00	Increased unsaturated water flux at the repository	NFE Include (primary FEP. Climate change is included) Exclude – Low consequence (secondary FEP on water quenching hot waste package) UZ Include EBS Include

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.08.02.00	Enhanced influx (Philip's drip)	NFE Include UZ Include EBS Exclude – Low consequence
2.1.08.04.00	Condensation forms on backs of drifts	EBS Include
2.1.08.05.00	Flow through invert	EBS Include
2.1.08.06.00	Wicking in waste and EBS	EBS Include
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and EBS	WF Misc. Include (through the use of a series of linked one dimensional flowpaths and mixing cells through the EBS, drip shield, waste package and into the invert) Exclude – Low consequence (preferential pathways within the EBS, WF and invert based on beneficial consequence) EBS Include
2.1.08.08.00	Induced hydrological changes in the waste and EBS	WF Misc. Include (induced hydrological changes (flow areas) from corrosion for the waste package and drip shield, induced hydrological changes [exposed fuel area] for the WF) Exclude – Low consequence (changes to hydrological properties for the WF, changes to hydrological properties for the invert) EBS Include
2.1.08.09.00	Saturated groundwater flow in waste and EBS	EBS Exclude – Low consequence
2.1.08.11.00	Resaturation of repository	NFE Include EBS Include
2.1.08.12.00	Drainage with Transport - Sealing and Plugging	EBS Exclude – Low consequence
2.1.08.13.00	Drains	EBS Exclude – Low (Zero) Probability

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.09.01.00	Properties of the potential carrier plume in the waste and EBS	NFE Include WF Misc. Include (potential effects of carrier plume by evaluating the influence of steel corrosion on the water chemistry in order to establish an uncertainty band) Exclude – Low consequence (the changing properties of incoming water, as evaluated by EBS) EBS Include
2.1.09.02.00	Interation with corrosion products	WF Misc. Include (the presence of a rind around the fuel pellets on the availability of water for radionuclides dissolution; the interaction between the expanding rind and the cladding, both in sealing of the gap and in unzipping the cladding; selected chemical effects in the integrated source term for each WF) Exclude (the potential effects from corrosion products on advective or diffusive transport of water and radionuclides; the potential sorptive effects from corrosion products [see YMP No. 2.1.09.05.00]) EBS Include
2.1.09.05.00	In-drift sorption Package	WF Misc. Include (sorption on mobile colloidal material) Exclude (sorption within the WF/WP based on beneficial consequence [conservative]) EBS Exclude – Low consequence
2.1.09.06.00	Reduction-oxidation potential in waste and EBS	WF Misc. Include EBS Include
2.1.09.07.00	Reaction kineticsÿin waste and EBS	WF Misc. Include (reaction kinetics in the equilibrium model) Exclude – Low consequence (reaction transients) EBS Exclude – Low consequence
2.1.09.08.00	Chemical gradients / enhanced diffusion in waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.09.11.00	Waste-rock contact	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.09.12.00	Rind (altered zone) formation in waste, EBS, and adjacent rock	NFE Included (in THC model) Excluded – Low consequence (from TH model) WF Misc. Included (in radionuclide mobilization). Excluded (in adjacent rock) EBS Include
2.1.09.13.00	Complexation by organics in waste and EBS	WF Misc. Exclude – Low probability EBS Exclude – Low consequence
2.1.09.14.00	Colloid formation in waste and EBS	WF Col Include EBS Include
2.1.09.15.00	Formation of true colloids in waste and EBS	WF Col Exclude (Note: this FEP addresses WFs only) EBS Exclude – Low consequence
2.1.09.16.00	Formation of pseudo-colloids (natural) in waste and EBS	WF Col Include EBS Include
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and EBS	WF Col Include EBS Include
2.1.09.18.00	Microbial colloid transport in the waste and EBS	WF Col Exclude EBS Exclude – Low consequence
2.1.09.19.00	Colloid transport and sorption in the waste and EBS	WF Col Exclude (in WF and waste package) EBS Exclude – Low consequence
2.1.09.20.00	Colloid filtration in the waste and EBS WF Col-Colloid filtration	WF Col Exclude EBS Exclude – Low consequence
2.1.09.21.00	Suspensions of particles larger than colloids	SZ Exclude – Low consequence WF Col Exclude EBS Exclude – Low consequence

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.10.01.00	Biological activity in waste and EBS	WP Include (waste container) Exclude – Low consequence (drip shield [TBV pending additional data and/or analysis]) WF Col Exclude EBS Include
2.1.11.01.00	Heat output / temperature in waste and EBS	NFE Include WF Misc. Include EBS Include
2.1.11.03.00	Exothermic reactions in waste and EBS Exothermic. Exothermic Reactions and Other Thermal Effects in Waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.11.04.00	Temperature effects / coupled processes in waste and EBS	WF Misc. Include EBS Include
2.1.11.05.00	Differing thermal expansion of repository components	WP Exclude – Low consequence (TBV pending additional inputs and/or analysis from Waste Package Design) WF Misc. Include EBS Exclude – Low consequence
2.1.11.07.00	Thermally-induced stress changes in waste and EBS	WF Misc. Include (thermally induced stress changes in NFE) Exclude – Low consequence (thermally induced stress changes in the waste and packaging based on low consequence [by design]) EBS Include
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and EBS	WF Misc. Include EBS Include

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS	WF Misc. Include (the amount of water reaching the EBS and eventually the waste is based on hydrologic calculations that consider the effects of temperature) Exclude – Low consequence (two-phase flow within the waste, thermally driven single-phase flow within the waste) EBS Include
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.01.00	Gas generation	UZ Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.02.00	Gas generation (He) from fuel decay	WF Misc. Exclude – Low consequence (the effects of He gas generation from performance-assessment calculations on the basis of low consequence to behavior of the waste package as part of overall repository performance assessment) EBS Exclude – Low consequence
2.1.12.03.00	Gas generation (H2) from metal corrosion	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.05.00	Gas generation from concrete	EBS Exclude – Low consequence
2.1.12.06.00	Gas transport in waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.12.07.00	Radioactive gases in waste and EBS	WF Misc. Exclude – Low consequence (the effects of noble [Ar, He, Kr, Rn, Xe] and CO_2 and CH_4 gas generation on the basis of low consequence to waste package behavior within the overall repository performance assessment) EBS Exclude – Low consequence
2.1.12.08.00	Gas explosions	WF Misc. Exclude – Low probability EBS Exclude –Low consequence
2.1.13.01.00	Radiolysis	WP Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude
2.1.13.02.00	Radiation damage in waste and EBS	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence (the effects of radiolysis-enhanced dissolution of spent nuclear fuel on the basis of low consequence to the performance of the disposal system during the regulatory timeframe) EBS Exclude – Low consequence (backfill, seals, pedestal, etc.)
2.1.13.03.00	Mutation	WF Col Exclude WF Misc. Exclude – Low consequence EBS Exclude – Low consequences
2.2.07.06.00	Episodic / pulse release from repository	UZ Exclude – Low consequence (effects of episodic flow resulting from episodic infiltration, effects of transient flow due to thermal-hydrologic processes on radionuclide transport) Include (effects of transient flow due to thermal-hydrologic processes on drift seepage, effects of intermittent waste package failures) EBS Include
2.2.08.04.00	Redissolution of precipitates directs more corrosive fluids to containers	NFE Include UZ Include EBS Include

Table B-12. Screening Decisions and Bases for Engineered Barrier System Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.11.02.00	Gas pressure effects	UZ Exclude – Low consequence and low probability EBS Exclude – Low consequence

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-13. Screening Decisions and Bases for Waste Package Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.1.03.01.00	Error in waste or backfill emplacement	WP Exclude – Low probability EBS Exclude – By regulation
1.2.02.03.00	Fault movement shears waste container	WP Exclude – Low probability DE Exclude – Low probability
1.2.03.02.00	Seismic vibration causes container failure	WP Exclude (TBV) – Low consequence DE Exclude (TBV) – Low consequence (waste package) Include (drip shield and fuel-rod cladding)
1.2.04.04.00	Magma interacts with waste	WP Include DE Include WF Misc. Include
2.1.03.01.00	Corrosion of waste containers	WP Include EBS Include
2.1.03.02.00	Stress corrosion cracking of waste containers and drip shields	WP Include (waste container) Exclude – Low consequence (drip shield)
2.1.03.03.00	Pitting of waste containers and drip shields	WP Include

Table B-13. Screening Decisions and Bases for Waste Package Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.03.04.00	Hydride cracking of waste containers and drip shields	WP Exclude – Low consequence (drip shield) Exclude – Low probability (waste container TBV pending additional data and/or analysis). Low probability (for waste package outer barrier)
2.1.03.05.00	Microbially-mediated corrosion of waste container and drip shield	WP Include (waste container) Exclude – Low consequence (drip shield TBV pending additional data and/or analysis)
2.1.03.06.00	Internal corrosion of waste container	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence (prior to waste package breach Include (after waste package breach)
2.1.03.07.00	Mechanical impact on waste container and drip shield	WP Exclude – Low consequence (mechanical damage of the waste container and drip shield by rock fall, mechanical damage of the waste container and drip shield by ground motion during seismic events [TBV pending additional data and/or analysis], mechanical damage by internal gas pressure and swelling corrosion products)
2.1.03.08.00	Juvenile and early failure of waste containers and drip shields	WP Include (manufacturing and welding defects in waste container degradation analysis) Exclude – Low consequence (manufacturing defects in drip shield degradation analysis) Exclude – Low consequence (early failure of waste container and drip shield from improper quality control during the emplacement)
2.1.03.09.00	Copper corrosion	WP Exclude – Low probability
2.1.03.10.00	Container healing	WP Exclude – Low consequence EBS Include
2.1.03.11.00	Container form	WP Exclude – Low consequence
2.1.03.12.00	Container failure (long-term)	WP Include EBS Include
2.1.06.06.00	Effects and degradation of drip shield	WP Exclude – Low consequence (damage to drip shield by rock fall, damage to drip shield by ground motion during seismic events [TBV pending additional data and/or analysis]) Include (physical and chemical degradation processes) EBS Include

Table B-13. Screening Decisions and Bases for Waste Package Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.06.07.00	Effects at material interfaces	WP Include EBS Exclude – Low consequence
2.1.07.01.00	Rockfall (large block) WF Clad-Rockfall	WP Exclude – Low consequence DE Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.07.05.00	Creeping of metallic materials in the EBS	WP Exclude – Low consequence (TBV pending additional inputs and/or analysis from Waste Package Design) EBS Exclude – Low consequence
2.1.09.03.00	Volume increase of corrosion products	WP Exclude – Low consequence WF Misc. Include (clad unzipping due to wet oxidation of CSNF) Excluded – Low consequence (dry oxidation of CSNF)
2.1.09.09.00	Electrochemical effects (electrophoresis, galvanic coupling) in waste and EBS WP-Electrochemical Effects in Waste and EBS	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence
2.1.10.01.00	Biological activity in waste and EBS	WP Include (waste container) Exclude – Low consequence (drip shield (TBV pending additional data and/or analysis)) WF Col Exclude EBS Include
2.1.11.05.00	Differing thermal expansion of repository components	WP Exclude – Low consequence (TBV pending additional inputs and/or analysis from Waste Package Design) WF Misc. Include EBS Exclude – Low consequence

Table B-13. Screening Decisions and Bases for Waste Package Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.11.06.00	Thermal sensitization of waste containers and drip shields increases their fragility	WP Include
2.1.12.03.00	Gas generation (H2) from metal corrosion	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.13.01.00	Radiolysis	WP Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude
2.1.13.02.00	Radiation damage in waste and EBS	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence (the effects of radiolysis-enhanced dissolution of spent nuclear fuel on the basis of low consequence to the performance of the disposal system during the regulatory timeframe) EBS Exclude – Low consequence (backfill, seals, pedestal, etc.)

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.04.04.00	Magma interacts with waste	WP Include DE Include WF Misc. Include
2.1.01.01.00	Waste inventory	WF Misc. Include; however, only a limited number of radionuclides are shown to be important to repository performance.
2.1.01.02.00	Codisposal / co-location of waste	WF Misc. Include (co-location and codisposal, chemical interactivity between DSNF and high-level radioactive waste-glass affects both DSNF degradation and radionuclide mobilization) Exclude – no credit for DSNF cladding or for any beneficial effects of DSNF and glass-pour canisters as barriers to DSNF degradation, to high-level radioactive waste-glass dissolution, or to radionuclide release. Exclude – Low consequence (DSNF geometry area dependence, dependence of radionuclide release on DSNF surface area) Exclude – Low probability (chemical interactivity between waste packages) Exclude – Low probability or Low consequence (preferential condensation based on (1) thermal shielding caused by the near-field averaging of the thermal field renders preferential condensation a process of low consequence and (2) occurrence of repository condensation and added uncertainty regarding the occurrence of preferential condensation render preferential condensation a process of low probability)
2.1.01.03.00	Heterogeneity of WFs	WF Misc. Include
2.1.01.04.00	Spatial Heterogeneity of Emplaced Waste	WF Misc. Exclude – Low consequence
2.1.02.01.00	DSNF degradation, alteration, and dissolution	WF Misc. Include
2.1.02.02.00	CSNF alteration, dissolution, and radionuclide release	WF Misc. Include (See other FEPs on specific phenomenon included and excluded)
2.1.02.03.00	Glass Degradation, Alteration, and Dissolution	WF Misc. Include (in package chemistry-dependent corrosion rates and congruent dissolution) Exclude – Low probability [credibility] (phase separation) Exclude – Low consequence (selective leaching) Exclude – Conservatively bounded (precipitation of silicate and other minerals)
2.1.02.04.00	Alpha recoil enhances dissolution	WF Misc. Exclude – Low consequence (effects of alpha-recoil from performance-assessment calculations on the basis of low consequence to the performance of the disposal system)
2.1.02.05.00	Glass cracking and surface area	WF Misc. Include

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.02.06.00	Glass recrystallization	WF Misc. Exclude
2.1.02.07.00	Gap and grain release of Cs, I	WF Misc. Include (gap and grain-boundary inventory produced while in repository) Exclude (additional gap and grain-boundary inventory potentially produced while in repository, and any reactions which would mitigate the gap and grain-boundary inventory and, thereby, releases)
2.1.02.08.00	Pyrophoricity	WF Misc. Exclude – Low consequence
2.1.02.09.00	Void space (in glass container)	WF Misc. Include (concept of unfilled void volume in TSPA-SR/License Application calculations)
2.1.02.10.00	Cellulosic degradation	WF Misc. Exclude – Low probability
2.1.02.11.00	Waterlogged rods	WF Clad Exclude – Low consequence
2.1.02.12.00	Cladding degradation before YMP receives it	WF Clad Include
2.1.02.13.00	General corrosion of cladding	WF Clad Exclude – Low consequence
2.1.02.14.00	Microbial corrosion of cladding WF Clad-Microbiologically Influenced Corrosion of Cladding	WF Clad Exclude – Low probability
2.1.02.15.00	Acid corrosion of cladding from radiolysis	WF Clad Exclude – Low probability
2.1.02.16.00	Localized corrosion (pitting) of cladding	WF Clad Exclude – Low probability
2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding	WF Clad Exclude – Low consequence
2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding	WF Clad Exclude – Low consequence
2.1.02.19.00	Creep rupture of cladding	WF Clad Include
2.1.02.20.00	Pressurization from He production causes cladding failure	WF Clad Include
2.1.02.21.00	Stress corrosion cracking of cladding	WF Clad Include
2.1.02.22.00	Hydride embrittlement of cladding	WF Clad Exclude – Low probability

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.02.23.00	Cladding unzipping	WF Clad Exclude – Low probability (dry oxidation) Include (wet oxidation)
2.1.02.24.00	Mechanical failure of cladding	WF Clad Include
2.1.02.25.00	DSNF cladding degradation	WF Misc. Exclude – Low consequence and based on conservatism
2.1.02.26.00	Diffusion-controlled cavity growth WF Clad-Diffusion-Controlled Cavity Growth	WF Clad Exclude – Low probability Include (general creep rupture process, is included-see FEP 2.1.02.19.00)
2.1.02.27.00	Localized corrosion perforation from fluoride	WF Clad Include
2.1.03.06.00	Internal corrosion of waste container	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence (prior to waste package breach Include (after waste package breach)
2.1.07.01.00	Rockfall (large block) WF Clad-Rockfall	WP Exclude – Low consequence DE Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and EBS	WF Misc. Include (through the use of a series of linked one dimensional flowpaths and mixing cells through the EBS, drip shield, waste package and into the invert) Exclude – Low consequence (preferential pathways within the EBS, WF and invert based on beneficial consequence) EBS Include
2.1.08.08.00	Induced hydrological changes in the waste and EBS	WF Misc. Include (induced hydrological changes [flow areas] from corrosion for the waste package and drip shield, induced hydrological changes [exposed fuel area] for the WF) Exclude – Low consequence (changes to hydrological properties for the WF, changes to hydrological properties for the invert) EBS Include
2.1.08.10.00	Desaturation / dewatering of the repository	NFE Include WF Misc. Include

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.08.15.00	Waste-form and backfill consolidation	WF Misc. Excluded – Low consequence (or possible slight beneficial consequence which is conservatively ignored)
2.1.09.01.00	Properties of the potential carrier plume in the waste and EBS	NFE Include WF Misc. Include (potential effects of carrier plume by evaluating the influence of steel corrosion on the water chemistry in order to establish an uncertainty band) Exclude – Low consequence (the changing properties of incoming water, as evaluated by EBS) EBS Include
2.1.09.02.00	Interaction with corrosion products	WF Misc. Include (the presence of a rind around the fuel pellets on the availability of water for radionuclide dissolution; the interaction between the expanding rind and the cladding: both in sealing of the gap and in unzipping of the cladding; selected chemical effects in the integrated source term for each WF) Exclude (the potential effects from corrosion products on advective or diffusive transport of water and radionuclides; the potential sorptive effects from corrosion products [see YMP No. 2.1.09.05.00]) EBS Include
2.1.09.03.00	Volume increase of corrosion products	WP Exclude – Low consequence WF Misc. Include (clad unzipping due to wet oxidation of CSNF) Excluded – Low consequence (dry oxidation of CSNF)
2.1.09.04.00	Radionuclide solubility, solubility limits, and speciation in the WF and EBS	WF Misc. Include
2.1.09.05.00	In-drift sorption Package-In-Package Sorption	WF Misc. Include (sorption on mobile colloidal material) Exclude (sorption within the WF/waste package based on beneficial consequence [conservative]) EBS Exclude – Low consequence
2.1.09.06.00	Reduction-oxidation potential in waste and EBS	WF Misc. Include EBS Include
2.1.09.07.00	Reaction kineticsÿin waste and EBS	WF Misc. Include (reaction kinetics in the equilibrium model) Exclude – Low consequence (reaction transients) EBS Exclude – Low consequence

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.09.08.00	Chemical gradients / enhanced diffusion in waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.09.09.00	Electrochemical effects (electrophoresis, galvanic coupling) in waste and EBS WP-Electrochemical Effects in Waste and EBS	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence
2.1.09.10.00	Secondary phase effects on dissolved radionuclide concentrations at the WF	WF Misc. Exclude – Low probability (Low probability due to uncertainty in amount of radionuclide actually being chemically bound, reasonably conclude complete release of radionuclides)
2.1.09.11.00	Waste-rock contact	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.09.12.00	Rind (altered zone) formation in waste, EBS, and adjacent rock	NFE Included (in THC model) Excluded – Low consequence (from TH model) WF Misc. Included (in radionuclide mobilization) Excluded (in adjacent rock) EBS Include
2.1.09.13.00	Complexation by organics in waste and EBS	WF Misc. Exclude – Low probability EBS Exclude – Low consequence
2.1.09.14.00	Colloid formation in waste and EBS	WF Col Include EBS Include
2.1.09.15.00	Formation of true colloids in waste and EBS	WF Col Exclude (Note: this FEP addresses WFs only) EBS Exclude – Low consequence
2.1.09.16.00	Formation of pseudo-colloids (natural) in waste and EBS	WF Col Include EBS Include
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and EBS	WF Col Include EBS Include

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.09.18.00	Microbial colloid transport in the waste and EBS	WF Col Exclude EBS Exclude – Low consequence
2.1.09.19.00	Colloid transport and sorption in the waste and EBS	WF Col Exclude (in WF and waste package) EBS Exclude – Low consequence
2.1.09.20.00	Colloid filtration in the waste and EBS WF Col-Colloid filtration	WF Col Exclude EBS Exclude – Low consequence
2.1.09.21.00	Suspensions of particles larger than colloids	SZ Exclude – Low consequence WF Col Exclude EBS Exclude – Low consequence
2.1.09.22.00	Colloid Sorption at the Air-Water Interface	WF Col Exclude
2.1.09.23.00	Colloidal stability and concentration dependence on aqueous chemistry	WF Col Include
2.1.09.24.00	Colloidal diffusion	WF Col Include
2.1.09.25.00	Colloidal phases are produced by coprecipitation (in waste and EBS)	WF Col Include (Note: only production of colloid phases by co-precipitation in the WP is considered here)
2.1.10.01.00	Biological activity in waste and EBS	WP Include (waste container) Exclude – Low consequence (drip shield TBV pending additional data and/or analysis) WF Col Exclude EBS Include
2.1.11.01.00	Heat output / temperature in waste and EBS	NFE Include WF Misc. Include EBS Include

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.11.03.00	Exothermic reactions in waste and EBS WF Misc. – Exothermic Reactions and Other Thermal Effects in Waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.11.04.00	Temperature effects / coupled processes in waste and EBS	WF Misc. Include EBS Include
2.1.11.05.00	Differing thermal expansion of repository components	WP Exclude – Low consequence (TBV pending additional inputs and/or analysis from Waste Package Design) WF Misc. Include EBS Exclude – Low consequence
2.1.11.07.00	Thermally-induced stress changes in waste and EBS	WF Misc. Include (thermally induced stress changes in NFE) Exclude – Low consequence (thermally induced stress changes in the waste and packaging based on low consequence by design) EBS Include
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and EBS	WF Misc. Include EBS Include
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and EBS	WF Misc. Include (the amount of water reaching the EBS and eventually the waste is based on hydrologic calculations that consider the effects of temperature) Exclude – Low consequence (two-phase flow within the waste, thermally driven single-phase flow within the waste) EBS Include
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.01.00	Gas generation	UZ Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.12.02.00	Gas generation (He) from fuel decay	WF Misc. Exclude – Low consequence (the effects of He gas generation from performance-assessment calculations on the basis of low consequence to behavior of the waste package as part of overall repository performance assessment) EBS Exclude – Low consequence
2.1.12.03.00	Gas generation (H2) from metal corrosion	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.06.00	Gas transport in waste and EBS	WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.12.07.00	Radioactive gases in waste and EBS	WF Misc. Exclude – Low consequence (the effects of noble [Ar, He, Kr, Rn, Xe] and CO_2 and CH_4 gas generation on the basis of low consequence to waste package behavior within the overall repository performance assessment) EBS Exclude – Low consequence
2.1.12.08.00	Gas explosions	WF Misc. Exclude – Low probability EBS Exclude –Low consequence
2.1.13.01.00	Radiolysis	WP Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude

Table B-14. Screening Decisions and Bases for Waste Form Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.1.13.02.00	Radiation damage in waste and EBS	WP Exclude – Low consequence (TBV pending additional data and/or analysis) WF Misc. Exclude – Low consequence (the effects of radiolysis-enhanced dissolution of spent nuclear fuel on the basis of low consequence to the performance of the disposal system during the regulatory timeframe) EBS Exclude – Low consequence (backfill, seals, pedestal, etc.)
2.1.13.03.00	Mutation	WF Col Exclude WF Misc. Exclude – Low consequence EBS Exclude – Low consequences
2.2.08.12.00	Use of J-13 well water as a surrogate for water flowing into the EBS and waste	WF Misc. Include
3.1.01.01.00	Radioactive decay and ingrowth	UZ Include SZ Include WF Misc. Include
3.2.07.01.00	Isotopic dilution	SZ Exclude – Low consequence WF Misc. Include (in waste package) Exclude – Low consequence (outside waste package, excluding isotopic dilution is conservative bounding)

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-15. Screening Decisions and Bases for Saturated Zone Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.02.01.00	Fractures	NFE Include (seepage) Exclude (permanent effects) UZ Include (effects of present-day fracture system) Exclude – Low consequence (effects of changes to the fracture system) SZ Exclude – Low consequence DE Include (existing fracture characteristics) Exclude – Low consequence (changes of fracture characteristics)
1.2.02.02.00	Faulting	UZ Include (effects of present-day faults) Exclude – Low consequence (the effects of changes to the faults) SZ Exclude – Low consequence DE Include (existing fault characteristics) Exclude – Low consequence (changes of fault characteristics)
1.2.03.01.00	Seismic activity	UZ Exclude – Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence (indirect effects) Exclude TBV – Low consequence (waste package) Include (drip shield damage and cladding damage)
1.2.06.00.00	Hydrothermal activity	UZ Exclude – Low probability SZ Exclude – Low consequence
1.2.09.02.00	Large-scale dissolution	UZ Exclude – Low probability SZ Exclude – Low consequence
1.2.10.01.00	Hydrological response to seismic activity	UZ Exclude – Low probability and Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence
1.3.07.01.00	Drought / water table decline	UZ Exclude – Low consequence SZ Exclude – Low consequence

Table B-15. Screening Decisions and Bases for Saturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.3.07.02.00	Water table rise	UZ Include SZ Include (changes in flux) Exclude – Assumed Low consequence (other effects not included in SZ Flow and Transport)
1.4.07.01.00	Water management activities	Bio Exclude – By regulation SZ Include
1.4.07.02.00	Wells	Bio Include (wells for human and agricultural use) Exclude – By regulation (wells located at a point other than specified by Revised Interim Guidance [RIG]) SZ Include
2.1.09.21.00	Suspensions of particles larger than colloids	SZ Exclude – Low consequence WF Col Exclude EBS Exclude – Low consequence
2.2.03.01.00	Stratigraphy	UZ Include SZ Include
2.2.03.02.00	Rock properties of host rock and other units	UZ Include SZ Include
2.2.06.02.00	Changes in stress (due to thermal, seismic, or tectonic effects) produce change in permeability of faults	UZ Exclude – Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence
2.2.06.03.00	Changes in stress (due to seismic or tectonic effects) alter perched water zones	UZ Exclude – Low consequence (effects of perched water changes below the potential repository) Include (effects of perched water changes above potential repository on drift seepage) SZ Exclude – Low consequence DE Exclude – Low consequence
2.2.07.12.00	Saturated groundwater flow	SZ Include

Table B-15. Screening Decisions and Bases for Saturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.07.13.00	Water-conducting features in the SZ-Water-Conducting Features	SZ Include
2.2.07.14.00	Density effects on groundwater flow	SZ Exclude – Low consequence
2.2.07.15.00	Advection and dispersion	SZ Include
2.2.07.16.00	Dilution of radionuclides in groundwater	SZ Include
2.2.07.17.00	Diffusion in the SZ	SZ Include
2.2.08.01.00	Groundwater chemistry / composition in UZ and SZ SZ-Groundwater Chemistry FEPs	UZ Include (effects of ambient-condition geochemistry) Exclude – Low consequence (changes in geochemical conditions) SZ Include
2.2.08.02.00	Radionuclide transport occurs in a carrier plume in geosphere SZ-Radionuclide Transport in a Carrier Plume	UZ Exclude – Low consequence SZ Include
2.2.08.03.00	Geochemical interactions in geosphere (dissolution, precipitation, weathering) and effects on radionuclide transport SZ-Groundwater Chemistry FEPs	NFE Include UZ Exclude – Low consequence SZ Include
2.2.08.06.00	Complexation in geosphere	UZ Include (effects of ambient-condition complexation) Exclude – Low consequence (effects of changes to complex formation due to changes in geochemical conditions) SZ Include
2.2.08.07.00	Radionuclide solubility limits in the geosphere	UZ Exclude – Low consequence SZ Exclude – Low consequence
2.2.08.08.00	Matrix diffusion in geosphere SZ-Matrix Diffusion	UZ Include SZ Include
2.2.08.09.00	Sorption in UZ and SZ	UZ Include SZ Include

Table B-15. Screening Decisions and Bases for Saturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.08.10.00	Colloidal transport in geosphere	UZ Include SZ Include
2.2.08.11.00	Distribution and release of nuclides from the geosphere SZ-Distribution and Release Of Nuclides	SZ Include
2.2.09.01.00	Microbial activity in geosphere SZ-Groundwater Chemistry FEPs	UZ Exclude – Low consequence SZ Include
2.2.10.01.00	Repository-induced thermal effects in geosphere	UZ Exclude – Low consequence (mountain-scale thermo-chemical effects) Include (thermo-chemical effects on drift seepage) SZ Exclude – Assumed low consequence or probability
2.2.10.02.00	Thermal convection cell develops in SZ	SZ Exclude – Low consequence
2.2.10.03.00	Natural geothermal effects SZ-Geothermal Effects	UZ Include SZ Include
2.2.10.06.00	Thermo-chemical alteration (solubility, speciation, phase changes, precipitation/dissolution) SZ-Groundwater Chemistry FEPs	NFE Exclude – Low consequence (except for the in-drift geochemical model that uses water chemistry and gas-phase composition from the drift-scale THC model that includes thermal-chemical alteration) UZ Exclude – Low consequence SZ Include
2.2.10.07.00	Thermo-chemical alteration of the Calico Hills unit SZ-Repository Induced Thermal Effects	UZ Exclude – Low consequence SZ Exclude – Assumed low consequence or probability
2.2.10.08.00	Thermo-chemical alteration of the SZ SZ-Repository Induced Thermal Effects	SZ Exclude – Assumed low consequence or probability
2.2.10.13.00	Density-driven groundwater flow (thermal) SZ-Repository Induced Thermal Effects	NFE Include SZ Exclude – Assumed low consequence or probability

Table B-15. Screening Decisions and Bases for Saturated Zone Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.11.01.00	Naturally-occurring gases in geosphere	UZ Exclude – Low consequence and low probability SZ Exclude – Low consequence
2.2.12.00.00	Undetected features (in geosphere) SZ-Undetected features	UZ Exclude – Low consequence and low probability SZ Include
2.3.02.02.00	Radionuclide accumulation in soils	Bio Include (Deposition) Exclude – By regulation (upwelling at other locations) SZ Exclude – Low consequence
2.3.11.04.00	Groundwater discharge to surface	SZ Exclude – Low consequence
3.1.01.01.00	Radioactive decay and ingrowth	UZ Include SZ Include WF Misc. Include
	Isotopic dilution	SZ Exclude – Low consequence WF Misc. Include (in waste package) Exclude – Low consequence (outside waste package, excluding isotopic dilution is conservative bounding)

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-16. Screening Decisions and Bases for Biosphere Features, Events, and Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.07.01.00	Erosion/denudation	UZ Exclude – Low consequence Bio Exclude – By regulation
1.2.07.02.00	Deposition	UZ Exclude – Low consequence Bio Exclude – By regulation
1.3.01.00.00	Climate change, global	UZ Include Bio Exclude – By regulation
1.3.04.00.00	Periglacial effects	UZ Exclude – Low probability Bio Exclude – By regulation
1.3.05.00.00	Glacial and ice sheet effects, local	UZ Exclude – Low probability Bio Exclude – By regulation
1.4.01.00.00	Human influences on climate	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.01.02.00	Greenhouse gas effects	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.01.03.00	Acid rain	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.01.04.00	Ozone layer failure	UZ Exclude – Low consequence Bio Exclude – By regulation
1.4.06.01.00	Altered soil or surface water chemistry	UZ Exclude – Low probability Bio Exclude – By regulation
1.4.07.01.00	Water management activities	Bio Exclude – By regulation SZ Include

Table B-16. Screening Decisions and Bases for Biosphere Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.4.07.02.00	Wells	Bio Include (wells for human and agricultural use) Exclude – By regulation (wells located at a point other than specified by RIG) SZ Include
1.4.08.00.00	Social and institutional developments	Bio Exclude – By regulation
1.4.09.00.00	Technological developments	Bio Exclude – By regulation
1.5.02.00.00	Species evolution	Bio Exclude – By regulation
2.2.07.03.00	Capillary rise	UZ Include Bio Exclude – By regulation
2.3.02.01.00	Soil type	Bio Include (soil type) Exclude – By regulation (soil development/formation)
2.3.02.02.00	Radionuclide accumulation in soils	Bio Include (Deposition) Exclude – By regulation (upwelling at other locations) SZ Exclude – Low consequence
2.3.02.03.00	Soil and sediment transport	Bio Include (aeolian) Exclude – By regulation (fluvial, glacial, bioturbation)
2.3.04.01.00	Surface water transport and mixing	Bio Exclude – By regulation
2.3.06.00.00	Marine features	Bio Exclude – By regulation
2.3.09.01.00	Animal burrowing / intrusion	Bio Exclude – By regulation
2.3.11.01.00	Precipitation	UZ Include Bio Include (precipitation) Exclude – By regulation (recharge/and climate change)
2.3.11.02.00	Surface runoff and flooding	UZ Include Bio Include (dispersion of contaminants, precipitation, and infiltration) Exclude – By regulation (recharge, water balance)
2.3.13.01.00	Biosphere characteristics	Bio Include (biosphere characteristics) Exclude – By regulation (conditions vary over time)

Table B-16. Screening Decisions and Bases for Biosphere Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.3.13.02.00	Biosphere transport	Bio Include (transport & transfer through biosphere compartments) Exclude – Low consequence and By regulation (radionuclide transfer to the biosphere via sediments, surface water, gas, and time dependent as well as chemical environment changes)
2.4.01.00.00	Human characteristics (physiology, metabolism)	Bio Include (adult) Exclude – By regulation (non-adult)
2.4.03.00.00	Diet and fluid intake	Bio Include (diet, fluids, and intakes other than drugs) Exclude – Low consequence and By regulation (filtration of water and food preparation and intake of drugs)
2.4.04.01.00	Human lifestyle	Bio Include (human lifestyle) Exclude – By regulation (hunter gathering)
2.4.07.00.00	Dwellings	Bio Include (household activities) Exclude – By regulation (location, building material, gas and water leakage, and space heating)
2.4.08.00.00	Wild and natural land and water use	Bio Exclude – Low consequence
2.4.09.01.00	Agricultural land use and irrigation	Bio Include (traditional crop and greenhouse farming) Exclude – By regulation (hydroponic gardening, peat and leaf harvesting and the use of ashes and sewage sludge and fire)
2.4.09.02.00	Animal farms and fisheries	Bio Include
2.4.10.00.00	Urban and industrial land and water use	Bio Exclude – By regulation
3.3.01.00.00	Drinking water, foodstuffs and drugs, contaminant concentrations	Bio Include (food stuff and water) Exclude – By regulation (drugs, non-well water)
3.3.02.01.00	Plant uptake	Bio Include (radionuclide uptake) Exclude – By regulation (natural outfalls)
3.3.02.02.00	Animal uptake	Bio Include (consumption of locally produced meat and associated produce) Exclude – Low consequence and By regulation (animal grooming & fighting, consumption of carcasses as well as scavengers and predators)
3.3.02.03.00	Bioaccumulation	Bio Include
3.3.03.01.00	Contaminated non-food products and exposure	Bio Exclude

Table B-16. Screening Decisions and Bases for Biosphere Features, Events, and Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
3.3.04.01.00	Ingestion	Bio Include (consumption of food stuffs and water) Exclude – By regulation (charcoal production, smoking, and tree sap consumption)
3.3.04.02.00	Inhalation	Bio Include
3.3.04.03.00	External exposure	Bio Include (external exposure to penetrating ionizing radiation) Exclude – Low consequence (dermal sorption and injection)
3.3.05.01.00	Radiation doses	Bio Include (exposure rate/dose conversion factors) Exclude (WIPP-specific FEP)
3.3.06.00.00	Radiological toxicity /effects	Bio Exclude – By regulation
3.3.06.02.00	Sensitization to radiation	Bio Exclude – By regulation
3.3.07.00.00	Non-radiological toxicity/effects	Bio Exclude – By regulation
3.3.08.00.00	Radon and radon daughter exposure	Bio Exclude

Source: Table B-1

NOTES: [a]Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

Table B-17. Screening Decisions and Bases for Disruptive Events Features, Events, Processes

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.01.01.00	Tectonic activity - large scale	DE Exclude – Low consequence
1.2.02.01.00	Fractures	NFE Include (seepage) Exclude (permanent effects) UZ Include (effects of present-day fracture system) Exclude – Low consequence (effects of changes to the fracture system) SZ Exclude – Low consequence DE Include (existing fracture characteristics) Exclude – Low consequence (changes of fracture characteristics)

Table B-17. Screening Decisions and Bases for Disruptive Events Features, Events, Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.02.02.00	Faulting	UZ Include (effects of present-day faults) Exclude – Low consequence (the effects of changes to the faults) SZ Exclude – Low consequence DE Include (existing fault characteristics) Exclude – Low consequence (changes of fault characteristics)
1.2.02.03.00	Fault movement shears waste container	WP Exclude – Low probability DE Exclude – Low probability
1.2.03.01.00	Seismic activity	UZ Exclude – Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence (indirect effects) Exclude TBV – Low consequence (waste package) Include (drip shield damage and cladding damage)
1.2.03.02.00	Seismic vibration causes container failure	WP Exclude (TBV) – Low consequence DE Exclude (TBV) – Low consequence (waste package) Include (drip shield and fuel-rod cladding)
1.2.03.03.00	Seismicity associated with igneous activity	DE Exclude – Low consequence (most effects) Include (damage to drip shields and fuel-rod cladding)
1.2.04.01.00	Igneous activity	DE Include (eruptive and intrusive events) Exclude – Low consequence (indirect effects)
1.2.04.02.00	Igneous activity causes changes to rock properties	UZ Exclude – Low consequence DE Exclude – Low consequence
1.2.04.03.00	Igneous intrusion into repository	DE Include (as a dike rather than as a sill) EBS Exclude. N/A for EBS
1.2.04.04.00	Magma interacts with waste	WP Include DE Include WF Misc. Include

Table B-17. Screening Decisions and Bases for Disruptive Events Features, Events, Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
1.2.04.05.00	Magmatic transport of waste	DE Exclude – Low consequence (transport in liquid magma) Include (pyroclastic transport)
1.2.04.06.00	Basaltic cinder cone erupts through the repository	DE Include
1.2.04.07.00	Ashfall	DE Include (ash cloud and surface deposition) Exclude (pyroclastic flow)
1.2.10.01.00	Hydrological response to seismic activity	UZ Exclude – Low probability and Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence
1.2.10.02.00	Hydrologic response to igneous activity	UZ Exclude – Low consequence DE Exclude – Low consequence
2.1.07.01.00	Rockfall (large block) WF Clad-Rockfall	WP Exclude – Low consequence DE Exclude – Low consequence WF Misc. Exclude – Low consequence EBS Exclude – Low consequence
2.1.07.02.00	Mechanical degradation or collapse of drift	DE Exclude – Low consequence EBS Exclude – Low consequence
2.2.06.01.00	Changes in stress (due to thermal, seismic, or tectonic effects) change porosity and permeability of rock	NFE Exclude – Low consequence DE Exclude – Low consequence
2.2.06.02.00	Changes in stress (due to thermal, seismic, or tectonic effects) produce change in permeability of faults	UZ Exclude – Low consequence SZ Exclude – Low consequence DE Exclude – Low consequence

Table B-17. Screening Decisions and Bases for Disruptive Events Features, Events, Processes (Continued)

FEP Number	FEP Name	Screening Decision and Basis[a]
2.2.06.03.00	Changes in stress (due to seismic or tectonic effects) alter perched water zones	UZ Exclude – Low consequence (effects of perched water changes below the potential repository) Include (effects of perched water changes above potential repository on drift seepage) SZ Exclude – Low consequence DE Exclude – Low consequence

Source: Table B-1

NOTES: [a] Issues or items given in parentheses are components of the FEP that are included or excluded, as indicated. If no issues or items are given in parentheses then the entire FEP is included or excluded, as indicated.

B.6. EVALUATION OF RELEVANT TOTAL SYSTEM PERFORMANCE ASSESSMENT ISSUE RESOLUTION STATUS REPORT ACCEPTANCE CRITERIA

Acceptance criteria for FEPs identification, classification, and screening are provided in the TSPAI IRSR (NRC 2000 [149372], Sections 4.1.1.2 and 4.2). Specific criteria are discussed below.

SUBISSUE 1–TRANSPARENCY AND TRACEABILITY, FEATURES, EVENTS, AND PROCESSES IDENTIFICATION AND SCREENING

These acceptance criteria address the screening process by which FEPs were included or excluded and the relationship between relevant FEPs. The origins of all YMP FEPs are described in Section B.2 of this Appendix and tracked in the database for each FEP. The screening process by which FEPs were included or excluded from the TSPA is described in Section B.4. Additional details on the screening process are provided in the individual FEP AMRs listed in Table B-1.

Relationships between relevant FEPs are identified in several ways. Related FEPs are inherently grouped together in accordance with the NEA-based hierarchical numbering scheme (Section B.3.1). The tree directory functionality in the database allows database users to graphically view and identify these groupings. Related FEPs are also grouped according to subject area (using database field input AMR). Finally, in future revisions to the database, related FEPs will be able to be identified using a keyword-search, pull-down menu.

SUBISSUE 2–SCENARIO ANALYSIS, IDENTIFICATION OF INITIAL FEATURES, EVENTS, AND PROCESSES

These acceptance criteria address the comprehensiveness of the FEP list. The YMP FEP list was initially developed from a comprehensive list of FEPs from other international radioactive waste disposal programs (Section B.2.1) and was supplemented with additional YMP-specific FEPs from project literature, technical workshops, and reviews (Sections B.2.2 through B.2.4). These bottom-up compilations produced an extensive, wide-ranging set of FEPs with the potential to influence the potential repository performance.

The comprehensiveness of the YMP FEP list derives in part from the NEA-based database structure. The NEA structure comprises a comprehensive group of subject areas (i.e., headings) potentially relevant to radioactive waste disposal that was developed to systematically classify the FEPs from seven different international programs. Continuous iterative review (i.e., at workshops and in FEP AMRs) of all database subject areas assures a strong degree of comprehensiveness, and ensures that no subject area is overlooked. Further assurance of comprehensiveness arises from the results of the most recent iterative reviews (Tables B-5 and B-6). Only nine and two new FEPs, respectively, were identified, and these new FEPs were variants of existing FEPs rather than representing entirely new subject areas. The diminishing returns of these iterative reviews suggest that the REV00 YMP FEP list is quite comprehensive.

SUBISSUE 2–SCENARIO ANALYSIS, CLASSIFICATION OF FEATURES, EVENTS, AND PROCESSES

These acceptance criteria address the grouping and categorization of FEPs. The all-inclusive bottom-up approach used to develop the YMP FEP list resulted in considerable redundancy in the FEP list. To eliminate the redundancy and to create a more efficient aggregation of FEPs to carry forward into the screening process, each of the 1,797 entries catalogued in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D) was identified as either a primary, secondary, or classification (layer, category, or heading) entry. The process and criteria for assigning FEPs to one of these categories is described in Section B.3.2. Because any categorization of FEPs is subjective, the preliminary identification of primary, secondary, and classification entries was reviewed and, where necessary, revised by subject matter experts.

This categorization resulted in a list of 323 primary FEPs that were carried forward for screening. Screening of the secondary (and classification) FEPs was not required, because the aspects of the secondary FEPs were encompassed by the primary FEPs.

SUBISSUE 2–SCENARIO ANALYSIS, SCREENING OF FEATURES, EVENTS, AND PROCESSES

These acceptance criteria address the screening of the FEPs. The regulatory criteria for screening FEPs on the basis of low probability, low consequence, or regulatory specification are summarized in Section B.4.1. To satisfy these regulatory screening criteria and to satisfy the TSPAI IRSR FEP screening acceptance criteria, guidelines were established for the content of the screening discussions. However, in some cases the screening discussions input from the FEP AMRs did not fully satisfy the guidelines, and, consequently, the screening information catalogued in the YMP FEP Database REV 00 (CRWMS M&O 2000 [150806], Appendix D) for some FEPs does not completely address the acceptance criteria of this subissue. Subsequent revisions of the FEP AMRs are planned to fully address this subissue and will be reflected in subsequent revisions of the database.

B.7. SUMMARY

This Appendix provides an introduction to the YMP FEP Database REV 00 (CRWMS M&O 2000 [150806]) and a summary of the screening decisions and bases given in the database for primary FEPs. The database structure is hierarchical, consisting of overarching classification entries (levels, categories, and headings), primary FEPs, and secondary FEPs. The primary FEPs collectively capture all of the issues relevant to the postclosure performance of the potential Yucca Mountain repository. Each primary FEP requires a screening discussion identifying the technical basis for inclusion or exclusion of FEPs in the TSPA-SR analyses. Secondary FEPs are subsumed in or redundant to overlying primary FEPs and do not require screening discussions.

Full screening discussions (screening arguments or summaries of TSPA dispositions) are given in the YMP FEP Database REV00 (CRWMS M&O 2000 [150806], Appendix D). These screening discussions were prepared by subject matter experts and documented in FEP AMRs listed in Table B-1.

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APPENDIX C

NATURAL-ANALOGUE INVESTIGATIONS IN SUPPORT OF PERFORMANCE ASSESSMENT OF THE POTENTIAL YUCCA MOUNTAIN RADIOACTIVE-WASTE REPOSITORY

APPENDIX C

ACRONYMS AND ABBREVIATIONS

SZ	saturated zone
TSPA-VA	Total System Performance Assessment-Viability Assessment
UZ	unsaturated zone
YMP	Yucca Mountain Site Characterization Project

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APPENDIX C

NATURAL-ANALOGUE INVESTIGATIONS IN SUPPORT OF PERFORMANCE ASSESSMENT OF THE POTENTIAL YUCCA MOUNTAIN RADIOACTIVE-WASTE REPOSITORY

C.1 INTRODUCTION

Natural-analogue studies are useful in performance-assessment investigations of potential nuclear-waste repositories as a means to evaluate independently the various components of the system under evaluation. Natural-analogue investigations also attempt to provide confidence in estimated parameter values used to simulate the long-term behavior of the system under investigation. In practice, the goals of natural-analogue investigations are to find analogues for structures and materials planned for use in nuclear-waste disposal. Natural analogues refer to both natural and anthropogenic (human-induced) systems in which processes, similar to those that might be expected to occur in a radioactive-waste repository, have occurred over long time periods (decades to millennia) and over large spatial scales (up to tens of kilometers) that are not accessible to ordinary laboratory experiments. In addition, natural-analogue studies have attempted to characterize uranium ore deposits and their geologic/hydrologic/geochemical settings because they contain mineralogic assemblages similar to the mass of materials, both waste forms and waste containers, planned for geologic disposal at Yucca Mountain. The challenge in using natural-analogues to estimate parameters for performance-assessment studies is that the initial and boundary conditions of the analogues can be difficult to quantify accurately. Another challenge lies in the fact that, although natural geologic processes that may be analogous to those at potential nuclear-waste repositories have been active for millions of years, some analogous archaeological materials are only several thousand years old. In addition, many materials being considered for use in nuclear-waste disposal may not have analogues in nature, in the archaeological record, or in human history until very recently. However, despite their limitations, natural analogues can provide conservative bounding estimates of variable parameters and the natural progression of some geologic processes so that the performance of these natural analogues can provide enhanced confidence in the parameter set chosen for performance-assessment investigations.

The *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]) provides a comprehensive overview of past and ongoing natural-analogue investigations and describes the Yucca Mountain Project natural-analogue program. The *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]) describes the role of natural-analogues in providing an understanding of the unknown elements of the potential repository by comparing those elements with some known natural and anthropogenic elements. According to the *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917], p. 13.1-1), "The results of the analog (sic) studies will be used to corroborate estimates of the magnitude and limitations of operative processes, thus building realism into conceptual and numerical-process models that underlie performance assessment...". The *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]) places a strong emphasis on analogues to processes and discusses specific natural analogues under the categories:

- Waste form, engineered barrier system, and repository-materials analogues
- Natural-system analogues
- Total-system analogues such as Cigar Lake.

More direct emphases on natural-analogue investigations are presented in *Natural Analogues for the Unsaturated Zone* (CRWMS M&O 2000 [141407]) and in *Archaeological Analogues for Assessing the Long-Term Performance of a Mined Geologic Repository for High-Level Radioactive Waste* (Stuckless 2000 [151957]).

C.2 MISCELLANEOUS EXAMPLES OF NATURAL-ANALOGUES FOR MATERIALS AND PROCESSES

Several classes of natural analogues have been considered in the performance-assessment investigation of Yucca Mountain. The investigations of these analogues are in various stages of evaluation. The areas of investigation that are most relevant to the materials considered likely for use in the potential Yucca Mountain repository have included:

- **Metalliferous Materials**–The most important metals being considered for use in the potential Yucca Mountain repository are exotic alloys of nickel, chromium, iron, and titanium. In addition, stainless steel will be used for many components including the inner sleeve of the waste package developed to contain the waste materials. Natural analogues of these materials are not found in nature, and they are very different from historic iron and steel. Only meteorites are potential analogues of these materials (Johnson and Francis 1980 [125291]). However, the composition of most meteorites is not analogous to the current suite of alloys being considered for use in the potential Yucca Mountain repository. The values of metal-corrosion parameters used in the Yucca Mountain performance assessment are conservative estimates based on the evaluation of historic man-made iron products and laboratory experiments of the atmospheric corrosion of metals used in construction and in operating machines (Miller et al. 1994 [126089]).

- **Cementitious Materials**–Although the current design of the potential Yucca Mountain repository specifies only limited use of cementitious materials, it is possible that future design changes may include the use of various formulations of concrete for support and added strength in potential repository construction. As with metals, modern concretes, which rely on the formation of alkaline-earth silicates for strength and durability, are much different from historic concretes. Roman concretes, one possible analogue, are mainly calcareous and behave differently from modern Portland-type siliceous cements. However, modern concretes are similar to the natural-cement deposits at Maqarin in northern Jordan. At Maqarin, the hyperalkaline geochemical conditions in the area of these deposits tend to immobilize radionuclides and metals. Unfortunately, under oxidizing conditions, hyperalkaline groundwater is corrosive to metalliferous material. In the case of Yucca Mountain, the waste packages and supporting structures would probably experience accelerated corrosion if large amounts of cementitious material were used in the potential repository.

- **Ceramics**–Archaeological ceramics are quite long-lived, up to 25,000 years, and the mineralogic components of ceramics demonstrate durability in terms of millions of years. Some material scheduled for disposal at Yucca Mountain, such as plutonium, will be immobilized in a specialized, high-temperature ceramic. Although archaeological ceramics are long-lived, they were generally fired at lower temperatures and do not have the exotic chemical composition anticipated for the Yucca Mountain ceramics. However, despite their general applicability as analogues, archaeological ceramics display protective alteration products that enhance their resistance to weathering and to atmospheric corrosion. Whether this protective coating will form on the ceramic material that may be used for Yucca Mountain remains to be determined in laboratory investigations. Nonetheless, the observed long-term performance of low-fire archeological ceramics is encouraging with respect to the modern, high-fire ceramics which by analogy should have significantly longer performance periods.

- **Glass**–Archaeological and natural glasses, such as basaltic or volcanic glass, have long fascinated humankind, and glassmaking using similar technology has existed for thousands of years. Natural geologic glasses tend to be more stable than archaeological glass, especially in regard to changing environments relative to humidity. The borosilicate glass intended for use at Yucca Mountain is more similar in composition to basaltic glass than to archaeological glass. The glass to be used at Yucca Mountain will also be subject to irradiation from the fission products and actinides in the waste products. The long-term effects of this irradiation have no obvious natural analogue. Laboratory studies are still underway to determine some of the potential long-term properties of borosilicate glass.

- **Radionuclide Transport**–The evaluations of models, data sets, and assumptions that will be used to predict performance of the potential Yucca Mountain repository would benefit from the use of a comparison of predicted system performance with that of a natural analogue or analogues. Waste form degradation, waste mobilization, and radionuclide transport are long-term phenomena that are difficult to evaluate with a high degree of certainty, especially if limited to information gained through short-term laboratory and field studies. Natural formations and deposits that have existed for very long periods of time may reveal the operative analogous transport processes and help to determine which factors most affect these processes. The natural reactors at Oklo, Gabon, Equatorial Africa, which achieved criticality about 2.7 billion years ago, have been studied extensively for waste degradation, mobilization, and radionuclide transport (including plutonium) in both reducing and oxidizing environments (Jakubick and Church 1986 [100459], Curtis et al. 1989 [100438], Brookins 1990 [100387]; Chapuis and Blanc 1993 [127825]).

  The 2.7-million-year-old uranium deposit at Oklo has evidence that the relative abundance of ^{235}U in the deposit approached that of modern nuclear reactors (Gauthier-LaFaye et al. 1989 [124997]). The evidence indicates that natural fission reactions took place at this location, and it has been determined that at least 16 reactors were active over a period of about one-million years (Gauthier-LaFaye et al. 1989 [124997]). Oklo is believed to be the only naturally occurring example of critically produced transuranic elements (Gauthier-LaFaye et al. 1996 [125005]). Examination of the mineralogy of the deposit reveals several reactor cells that produced plutonium (Gauthier-LaFaye et al. 1996 [125005]). Past investigations at Oklo show that 3 elements–^{234}U, ^{239}Pu, and ^{242}Pu–do not appear to have migrated from the reactor-core uraninites, and that this behavior may also apply to ^{237}Np (Menet et al. 1992 [150921]). Other investigations at Oklo indicate that ^{99}Tc may have migrated both to inclusions in the minerals in the core of the natural reactor and to the clays that seal the ore deposit (Eberly et al. 1996 [150899]). The short half-lives of ^{14}C, ^{79}Se, and ^{231}Pa are probably the reason they are not prominently present in and around the Oklo reactors. Half-life may also be important in the case of ^{129}I which, with its ionic radius about double that of uranium, probably has migrated away from the Oklo reactors rather than being incorporated into the uraninite before decaying. The migration of ^{99}Tc movement to inclusions was detected by analyzing for its daughter product ^{99}Ru (Hidaka et al. 1993 [151769]).

Extensive research has been conducted on the Cigar Lake uranium deposit in northern Saskatchewan (Cramer and Smellie 1994 [100437]). Although located in saturated sandstone, important analogue information may still be applicable to Yucca Mountain. Significantly, although 1.3 billion years old and located in a fractured, saturated environment, there is no evidence of the deposit expressed at the surface. A 3-year analogue study was conducted at Pocos de Caldas, Brazil, which focused on radionuclide transport issues (Chapman et al. 1991 [100423]). Other potential analogue sites that have included studies of transport phenomena include Alligator Rivers in Australia, Tono in Japan, and Palmottu in Finland.

Peña Blanca, in Chihuahua, Mexico, is a uranium deposit located in unsaturated tuff and is an excellent analogue to the processes and material at the potential Yucca Mountain repository. Section C.3.2 provides a summary of a planned detailed investigation concerning whether or not future investigations and field studies of this uranium deposit may provide important insights into the process of radionuclide transport.

Natural Analogues for the Unsaturated Zone (CRWMS M&O 2000 [141407]) investigated the distribution of radioisotopes of uranium (U), radium, thorium, and protactinium in secondary minerals at the Nopal I mine site (Murrell et al. 1999 [130408]). The results of sample analyses indicate that uranium (^{235}U and ^{238}U), thorium, and protactinium are found in fracture filling materials in mineralogies that indicate long-term stability on the order of 100,000 years. However, radium appears to be mobile on the scale of 50 ka indicating that radium could be used as a natural conservative tracer.

- **Coupled Processes in the Unsaturated Zone**–*Natural Analogues for the Unsaturated Zone* (CRWMS M&O 2000 [141407]) summarizes the results of completed and ongoing field and laboratory investigations of natural analogues dealing with UZ flow and transport. The report concluded that natural analogues could be important in confidence building regarding the conceptual and numerical models used to evaluate the potential Yucca Mountain repository site.

The Idaho National Engineering and Environmental Laboratory conducted field investigations in unsaturated, fractured Quaternary basalt flows at Box Canyon, Idaho. A two-phase, dual-permeability model was used to simulate the flow of air and water through the fractured basalt. Pneumatic and fluid-infiltration tests provided data used to calibrate the model. Although somewhat scale dependent, the model analysis indicates that the infiltration established a vertically and laterally uniform front with no strong channeling along fractures (CRWMS M&O 2000 [141407]). Future application of this study could include conducting similar field investigations and modeling at Rainier Mesa at the Nevada Test Site to determine whether or not the Box Canyon natural analogue is relevant to the potential Yucca Mountain repository site.

Analogues to coupled thermal-hydrologic-mechanical-chemical processes include active geothermal systems and shallow magmatic intrusions into the UZ with associated hydrothermal processes. The examples of these processes are many and varied and not all are suitable analogues to the Yucca Mountain repository site. *Natural Analogues for the Unsaturated Zone* (CRWMS M&O 2000 [141407]) note that some locations such as Banco Bonito flow associated with the Valles Caldera in New Mexico indicate that the intrusion was associated with pore-water vaporization and reflux conditions such as those expected at Yucca Mountain. Alternatively, *Natural Analogues for the Unsaturated Zone* (CRWMS M&O 2000 [141407]) show that at the Grant's Ridge site west of Albuquerque, New Mexico, there is thermal alteration of minerals but little evidence of hydrothermal alteration. Future examination of active geothermal systems may provide insights regarding the nature of fracture flow during the condensate-drainage period after the potential post-repository-closure heat pulse.

C.3 NATURAL-ANALOGUE INVESTIGATIONS IN SUPPORT OF THE YUCCA MOUNTAIN TOTAL SYSTEM PERFORMANCE-ASSESSMENT

In addition to investigating natural analogues to materials intended for use in the potential Yucca Mountain repository, the results of some preliminary investigations of analogues to geologic processes are underway. Two investigations of interest are the simulation of the 1995 volcanic eruption at Cerro Negro, Nicaragua and a preliminary investigation of potential radionuclide transport at the Nopal I uranium deposit in the Sierra Peña Blanca in Chihuahua, northern Mexico. These two investigations will be discussed separately. The information and analysis is presented as corroborative information only, and does not require further confirmation.

C.3.1 SIMULATION OF 1995 CERRO NEGRO, NICARAGUA, VOLCANIC ERUPTION

Cerro Negro is a basaltic, cinder-cone volcano located 20-km northeast of the city of Leon in northwestern Nicaragua. Following a volcanic eruption with a significant ashfall plume at Cerro Negro in November 1995, the thickness of the volcanic ash deposited during that eruption was measured immediately after the eruption (Hill et al. 1998 [151040]). The ash-thickness data provided an opportunity to use the ASHPLUME code (CRWMS M&O 1999 [132547]) to simulate this eruption. The goal was to compare an ash-deposition calculation predicted using the ASHPLUME code to an actual ashfall event from a small-volume basaltic volcano. The essence of the analogy is that the Cerro Negro volcano is similar in size and type to a possible volcanic event that could occur in the vicinity of Yucca Mountain.

The two-dimensional diffusion model of Suzuki (1983 [100489]) was implemented by the ASHPLUME code. Two versions of the code were compared to the Cerro Negro ash-thickness data. Version 2.0 uses eruption power and event duration to determine column height and mass of ejecta, as described in Jarzemba et al. (1997 [100987]). Version 1.4LV uses the volume and density of the ash to determine ash column height and mass (CRWMS M&O 1999 [132547]). Input parameters used in the Cerro Negro comparison for each version are given in Table C-1. These parameters are given in Hill et al. (1998 [151040]) for the November 1995 Cerro Negro eruption.

Table C-1. ASHPLUME Parameters Used for Cerro Negro Comparison

Parameter	Version 1.4LV	Version 2.0
Ash density, g/cm^3	1.2	1.2
Particle shape factor	0.5	0.5
Air density, g/cm^3	0.001293	0.001293
Air viscosity, g/cm-s	0.00018	0.00018
Constant C, $cm^2/s^{5/2}$	400.0	400.0
Constant beta	10.0	10.0
Lower limit on column height, km	0.001	0.001
Mean ash particle diameter, cm	0.07	0.07
Particle diameter standard deviation	0.8	0.8
Wind speed, cm/s	900.0	900.0
Initial eruption velocity, cm/s	10000.0	10000.0
Eruption power, watts	NA	7.34×10^9
Event duration, s	NA	3.46×10^5
Eruption volume, km^3	0.000288	N/A

NOTE: N/A = not applicable

The version 2.0 simulation was completed first, and the initial eruption-velocity and particle-size standard-deviation parameters were adjusted to obtain a reasonable fit to the observed data. Next, the version 1.4LV simulation was performed using the same parameter set and a value for ash volume that the model used to produce a total mass of ash equivalent to the version 2.0 calculation.

The two versions of the code produce similar results. Figure C-1 shows the results from each code version compared to the measured Cerro Negro ash-deposit data. The ASHPLUME calculations compare well with the observed data for distances from the volcanic vent greater than 10 km. For distances less than 10 km, the ASHPLUME results give ash-thickness values greater than the observed data. The lobe on the northern side of mapped ash-plume data indicates that a variation in wind direction, speed, or both occurred during the eruption. This probably accounts for some of the discrepancy between the observed and the simulated data because ASHPLUME assumes a constant wind speed and direction for a given simulation. The results generally show that the ASHPLUME model can reasonably predict the ash-fall distribution of a basaltic cinder cone volcano similar to Cerro Negro. The Cerro Negro ash-fall-calculation method could be used to simulate possible ash-producing events either near the potential Yucca Mountain repository or an ash-producing eruption through the potential repository involving waste-package destruction and aerial liberation of radionuclides.

C.3.2 RADIONUCLIDE-TRANSPORT SIMULATIONS FOR THE PEÑA BLANCA NATURAL-ANALOGUE SITE

C.3.2.1 General Introduction

Although several natural geologic deposits and environments analogous to nuclear-waste repositories have been analyzed (Miller et al. 1994 [126089], Appendix A), most are not analogous in geologic age, geologic setting, climate, or mineralogy to the Yucca Mountain site. One of the most promising areas investigated vis-à-vis the Yucca Mountain site has been the Nopal I uranium mine, located in the Sierra Peña Blanca approximately 50 km (31 mi.) north of Chihuahua City, Chihuahua, Mexico, and usually referred to as the Peña Blanca natural analogue (Murphy 1995 [121310]).

The purpose of this study was to analyze the possible mobilization and groundwater transport of radionuclides potentially released from the Nopal I ore deposit. The goal of the investigation was to estimate whether further field investigations and groundwater sampling at the Nopal I site would provide a basis for a natural-analogue comparison of the Peña Blanca site with the expected performance of the Yucca Mountain site.

The Peña Blanca Nopal I uranium deposit is temporally, geologically, climatically, geochemically, and hydrologically analogous to Yucca Mountain (Murphy et al. 1991 [151772]). Previous studies at Peña Blanca have focused on the mineralogy of the uranium-bearing materials and on estimates of the source term for hydrochemical migration in the near-field area of the ore body (Pickett and Murphy 1997 [109989], Murphy and Pearcy 1992 [151773], Leslie et al. 1993 [101714], Green et al. 1995 [149528], Murphy 1995 [121310]). However, to date, the only performance-assessment analysis of Peña Blanca has been conducted by Murphy and Codell (1999 [149529]). Several performance-assessment analyses have simulated the performance of the potential Yucca Mountain deep geologic repository using the following processes:

- Precipitation input to the land surface above the potential repository
- Flow through the UZ
- Corrosion and degradation of the waste packages and waste forms
- Groundwater flow of contaminants through the saturated zone (SZ)
- The use of biosphere dose-conversion factors to estimate radionuclide concentrations at receptor wells downgradient from the potential repository.

These analyses used a numerical model to simulate the various processes involved in the performance of the potential repository and its engineered components. The model configuration generating the contaminants leaving the potential repository and subject to flow and transport of radionuclides began with the radionuclide inventory of spent nuclear fuel and other waste components and implemented dissolution models to these components to estimate the potential release of radionuclides at Peña Blanca.

The performance-assessment model for the Peña Blanca uranium deposit began with the basic assumption that the uranium oxide comprising the bulk of the radionuclides at the deposit is directly analogous to spent nuclear fuel, which is largely composed of uranium oxide. The Nopal I uranium deposit is approximately 9 million years old (Murphy 1992 [106833]). The starting point for the numerical simulations is approximately 3 million years ago, when the Nopal I deposit was exposed to oxidizing conditions (Murphy and Codell 1999 [149529]).

The following analysis used the Repository Integration Program (Golder Associates 1998 [100449]) to analyze potential release and transport of radionuclides from the Nopal I ore deposit in the Peña Blanca uranium district. Because of the limited data available for the site, the analysis is, of necessity, a scoping study that will require validation through additional field-data collection. Therefore, the performance-assessment analysis of the Peña Blanca site is accompanied by specific plans for additional investigations and data collection at the mine site.

C.3.2.2 Setting

The Peña Blanca Nopal I ore deposit lies in the high desert in the Basin and Range geologic province approximately 50 km (31 mi.) north of Chihuahua City, Chihuahua, Mexico (Figures C-2 and C-3) (Pearcy et al. 1993 [151774]). The ore body lies on the southeast side of the Sierra Peña Blanca, which is named for the white color of oxidized mineral deposits in the range. Field observations indicate that the original land surface at the Nopal I deposit probably exposed a small portion of the ore body. The deposit lies in fractured, welded, and altered rhyolitic ash-flow tuffs similar to the volcanic rocks at Yucca Mountain (George-Aniel et al. 1991 [105636]).

When the deposit was mined in the 1970s and 1980s, two prominent benches were cut across the ore body at the 0-m and +10- m levels (local, vertical mine coordinates). Two adits enter the mined area from land surface at the 0- m and 70- m levels, and the one principal shaft connects the 0- m adit with adits at the 20-, 40-, and 70- m levels, as shown on Figure C-4 (Reyes-Cortes 1997 [149533]). The total depth of the main mine shaft is approximately 110 m from the -20 m level.

The Nopal I ore deposit has been estimated to be approximately 9 million years old (Murphy 1992 [106833]). Examination of the weathering mineralogy of the ore body indicates that the deposit probably lay stable and under reducing conditions until approximately 3 million years ago, when it was exposed to oxidizing groundwater, infiltration from precipitation, and weathering processes (Murphy and Codell 1999 [149529]). Since that time, an altered zone of oxidized uranium and accessory minerals and secondary minerals has developed around the ore body (Murphy and Codell 1999 [149529]).

The geologic characterization of the Nopal I ore body is described in Goodell (1981 [149484]) and shown on Figure C-5. The ore deposit occurs in ignimbritic ash-flow tuffs similar to those found at the Yucca Mountain site (DOE 1998 [100550]). The top 30 m of the deposit lie in the

Nopal Formation, a fractured rhyolitic tuff, and, the lower 70 m lie in the Colorados Formation, a weakly welded, fractured, ignimbritic tuff. Beneath the deposit, and probably containing the lower several meters of the ore body, lie 60 m of Pozos Formation, a silicified, detrital conglomerate composed principally of altered limestone clasts, and approximately 10 m of Cretaceous limestone. A near vertical fault, with greater than 10 m of offset, intersects the hill containing the ore body, and lies to the east of the deposit, but does not appear to cut the ore body. Minor faults and fractures are observed in the vertical walls of the open faces of the mine above the 0- and 10-m levels.

The ore body can be approximately described as a roughly cylindrical, breccia-pipe-like form, approximately 18 by 30 m in the horizontal plane and 100 m in the vertical dimension. The deposit is presently estimated to contain 333 metric tons of uranium (George-Aniel et al. 1991 [105636]). Uranium comprises approximately 0.23 percent of the deposit by volume (George-Aniel et al. 1991 [105636]). Using analyses of oxidized uranium minerals (Pearcy et al. 1993 [151774]), Murphy and Codell (1999 [149529]) estimated that the unoxidized ore body contained 408 metric tons of uranium as uranium oxide.

Elements of the site hydrogeology are presented in Green et al. (1995 [149528]) and Pearcy et al. (1993 [151774]). The Nopal I ore deposit lies in the UZ, and the water table has been estimated to be approximately 70 m below the ore deposit, although the base of the deposit has not been firmly established (Pearcy et al. 1993 [151774]). Green and Rice (1995 [149485]) conducted an artificial recharge study at the Nopal I mine, and the data obtained were used to estimate the hydraulic properties of the intact and oxidized portions of the ore body. Their investigations indicated that the porosity of the ore deposit ranged from 0.05 to 0.08 in unaltered rock, and to 0.30 in altered rock. The saturated hydraulic conductivity ranged from 6×10^{-12} cm/s in the unaltered rock, and to 1×10^{-7} cm/s in the altered rock.

Previous studies of the Nopal I deposit have focused on the geochemical aspects of the oxidation, transport, and reprecipitation of uranium mineral phases in the matrix and bedrock (Pearcy et al. 1993 [151774], Pickett and Murphy 1997 [109989], Pearcy 1994 [149523]). The possible release and transport of likely conservative species such as ^{99}Tc has not been previously considered in detail. The following performance-assessment analysis considers the possibility of the generation, liberation, and transport of ^{99}Tc and uranium species from the Nopal I deposit.

C.3.3 MODEL ELEMENTS

C.3.3.1 Radionuclide Inventory

The estimated uranium content of the original, preoxidized Nopal I deposit was 408 metric tons (Murphy and Codell 1999 [149529]). The inventory of uranium species and some daughter products, together with radiochemical analyses of Nopal I ore piles and vegetation residues (Leslie et al. 1999 [109967]), were used to estimate the inventory of radionuclides used for the Repository Integration Program simulations as shown on Table C-2, in Section C.3.3.6 below. The inventory is approximate, and the Repository Integration Program analyses were conducted using a range of values surrounding these estimates. The ^{99}Tc inventory was based on information found in Curtis et al. (1999 [110987]) and is not specifically related to Nopal I. More detailed radiochemical analyses are anticipated to be obtained from core and/or chip samples from the planned drilling at the Nopal I site. These future analyses will provide a more definitive radionuclide inventory for future performance-assessment-type calculations.

C.3.3.2 Conceptual Model

The performance-assessment analysis of Nopal I assumed the Peña Blanca area to be located in an area where there is lateral to vertically downward groundwater flow (Green et al. 1995 [149528]). The site of the Nopal I mine is in a recharge condition relative to the regional groundwater flow (Green et al. 1995 [149528]). A significant playa, Playa Cuervo, exists east of the Sierra Peña Blanca and is assumed to be the regional discharge location for groundwater in the Peña Blanca area (Reyes-Cortes 1997 [149533]). The physiography of the area is similar to that of the Great Basin where long-term data on climate and precipitation patterns were assembled in preparation of the Total System Performance Assessment-Viability Assessment (TSPA-VA) (CRWMS M&O 1998 [100356]). Because of similarities in geography and physiography and the lack of local data for the state of Chihuahua, the use of the long-term infiltration patterns used in the TSPA-VA for Yucca Mountain are considered to be adequate for the scoping-type calculations for the Peña Blanca performance-assessment analysis.

The Nopal I deposit is approximately 9 million years old (Murphy 1992 [106833]), but the deposit was only exposed to oxidizing conditions over the last 3.2 to 3.4 million years (Pickett and Murphy 1997 [109989]). The Repository Integration Program model formulation considers a steady-state condition in which the source term depends on the degradation rate. The UZ and the SZ are divided into mixing cells to account for diffusion and dispersion.

The conceptual model for radionuclide transport assumes that ^{99}Tc has been generated primarily by the spontaneous fission of ^{238}U (Curtis et al. 1999 [110987]). Curtis et al. (1999 [110987]) also indicate that ^{99}Tc may also be generated by irradiation of molybdenum and, possibly, by neutron-induced fission of ^{235}U. However, there is not a strong molybdenum presence in the Peña Blanca mining district, and there is scant information on the generation of ^{99}Tc from ^{235}U at Nopal I (Reyes-Cortes 1997 [149533]). Therefore, the ^{99}Tc inventory for the Repository Integration Program simulations was developed based on the abundance of uranium species at the Nopal I deposit and used the estimation algorithm presented in Curtis et al. (1999 [110987]) to estimate possible ^{99}Tc production from spontaneous fission of ^{238}U.

The ^{99}Tc produced from ^{238}U is assumed to be subject to leaching from the ore body and to be largely conservative (Beasley et al. 1998 [102430]) as the soluble and mobile form of pertechnetate ion ($TcO4^-$) (Brown 1997 [151775]), and then available for transport by groundwater through the UZ and SZ. The work of the Southwest Research Institute indicates that uranium-mineral alteration at Nopal I has not led to significant migration of uraniferous species; rather, it has led to the precipitation of secondary uranium minerals and sorption of the liberated uranyl species after tens of meters of lateral travel distance (Pearcy et al. 1995 [110223]). Given the age of the Nopal I deposit, this performance assessment assumes that ^{99}Tc can be used as a surrogate radionuclide to estimate the potential transport of conservative species in the natural environment.

In the model, the Nopal I ore deposit was assumed to be accessed by infiltrating water, and it was assumed that the original and present uraninite in the ore body was subject to dissolution. The dissolution was followed by the release of uranium species in the same manner as the waste form in the waste packages in the performance-assessment modeling of the potential Yucca Mountain repository. However, in the Peña Blanca analysis, there was no waste package to delay water-uraninite interaction or to hinder direct release and transport. Also, all the dissolved species were subject to advective or to diffusive flux from the ore body to the UZ.

C.3.3.3 Infiltration

Precipitation infiltration has not been estimated at the Nopal I site, but the vegetation and soil of the area indicate that infiltration is likely (Leslie et al. 1999 [109967]). The climate of the Peña Blanca region is arid, with an estimated 250 mm/yr of precipitation (Pearcy et al. 1993 [151774]). The basic Peña Blanca performance-assessment model assumed that the present-day climate used in the TSPA-VA is the most likely climatic regime during the last 3 million years. The presence of perched water in shot holes drilled into the open bench of the +10 m level indicates that precipitation infiltration occurs in the area of the Nopal I mine (Picket and Murphy 1999 [110009]). Because of the climatic similarity of Sierra Peña Blanca and the Yucca Mountain site, and because there have been no detailed infiltration studies at the Nopal I mine site, the infiltration observations made at Yucca Mountain (CRWMS M&O 1998 [100356]) were used in the Peña Blanca performance assessment. The Peña Blanca modeling thus assumed an infiltration rate of 5 to 7 mm/yr (CRWMS M&O 1998 [100356]).

C.3.3.4 Unsaturated Zone

The vertical UZ section of rock below the ore body is approximately 70-m thick (Reyes-Cortes 1997 [149533]) and is composed both of the lower part of the volcanic tuff and volcanic conglomerate of the Pozos Formation and of the upper part of the Cretaceous limestone that underlies this region of northern Mexico. In the absence of sufficient data for Nopal I to create a sophisticated site-specific model, the potential for transport of ^{99}Tc through the UZ beneath the Nopal I deposit was modeled using mixing cells in the Repository Integration Program. The model assumed that infiltrating water contacts the ore body and that the volume of percolating water corresponds to that derived from the present-day climate. The cross-sectional area of the ore body (18 by 30 m) was used with the height of the deposit (100 m) (Pearcy et al. 1993 [151774]) and the porosity range of 7.5 to 30 percent of the rock (Green and Rice 1995 [149485]) to estimate the volume of fluid percolating through the deposit. Uranium-oxide degradation was assumed to be the operating mechanism for degradation and dissolution of the ore body. Therefore, the performance-assessment analysis performed with the Repository Integration Program model used the same uranium-oxide dissolution model used in the Repository Integration Program for uranium-oxide spent fuel in the Yucca Mountain Site Characterization Project (YMP) performance-assessment analyses.

C.3.3.5 Saturated Zone

In a manner similar to YMP performance assessment models, the Peña Blanca analysis accumulates the water flowing through the UZ and passes it to the SZ at the water table (CRWMS M&O 1998 [100364], CRWMS M&O 1998 [100365]). The fluid contains radionuclides leached from the ore body based on the inventory input into the model. The fluid at Peña Blanca contains primarily ^{99}Tc. The theoretically contaminated groundwater flows eastward in the SZ through the Cretaceous limestone according to the hydraulic gradient (Green et al. 1995 [149528]). The groundwater flow in the model is through a stream tube in the same manner as in the TSPA-VA model. Nuclide concentrations were calculated at various points along the stream tube at the estimated locations of proposed groundwater monitor/sampling wells approximately 150, 600, and 1,300 m downgradient from the ore body. The 1,300-m location is at the same distance as the location of the former mining-camp water supply well, approximately downgradient from the ore body. The one additional downgradient well that was considered is the Papalote ranch supply well, which is approximately 2-km downgradient from the ore body.

The Peña Blanca Repository Integration Program model did not calculate dose-to-receptor values because the goal of the investigation was to estimate concentrations in groundwater of ^{99}Tc and some uranium species at selected distances from the Nopal I mine. The calculations provided by the Repository Integration Program were compared to analyses of water samples collected at the mining-camp water-supply well and were presented in Pickett and Murphy (1999 [110009]). In that study, the uranium was estimated to be 1.69×10^{-4} mg/L.

C.3.3.6 Model Configuration

The performance-assessment model for the Peña Blanca natural analogue was configured based on the physical description of the Nopal I mine site found in Pearcy et al. (1993 [151774]) and Goodell (1981 [149484]). The physical description of the site was supplemented with field observations by author Saulnier, with additional information provided by Dr. I. Reyes-Cortes during a site visit on June 28 to 29, 1999 (Saulnier 1999 [151776]).

The source term was assumed to be an ovoid cylinder of uranium oxide based on the dimensions exposed at the cleared +10-m and 0-m levels of the mine. The source was defined using a single source-term group in the Repository Integration Program and was approximated as a single waste package. This methodology was used to allow definition of a dissolution rate for the uranium-oxide ore. The source was available for dissolution from time zero, the time when the ore body was exposed at or near land surface at 3 Ma. Ambient temperature was assumed, and all the remaining parameters for fuel-dissolution rate calculation correspond to those used in the YMP performance-assessment model.

The radionuclide inventory described in Table C-2 was used as the initial waste-package inventory for the Peña Blanca model. Because of its short half-life, ^{99}Tc was further defined with an accumulation function (Curtis et al. 1999 [110987]) based on a steady-state inventory of ^{238}U, which has a half-life of 4.5×10^9 years. Using the assumed inventory, the dissolved mass from the ore deposit was mixed in two mixing cells, with the volume set equal to the physical volume of the uranium deposit (i.e., the ovoid cylinder), taking into account the porosity (Green and Rice 1995 [149485]). The total flow out of the mixing cell was set equal to the infiltration flux times the cross-sectional area of the deposit.

Table C-2. Radionuclide Inventory for Peña Blanca Repository Integration Program Simulations

Radionuclide Inventory for Repository Integration Program		
Radionuclide	Ci	Grams
^{238}U	137.360	4.080E+08
^{235}U	6.394	2.955E+06
^{234}U	144.228	23096.624
^{231}Pa	6.489	137.336
^{232}Th	0.640	5506.678
^{230}Th	113.677	5.806E+06
^{99}Tc	1.088E-05	6.406E-04

DTN: MO0009MWDPEN01.009 [153039]

The 70-m thick UZ was discretized into five mixing cells. The mass release from the source mixing cell (i.e., the ore deposit) was fed into the first UZ mixing cell at a rate equal to the infiltration flux times the cross-sectional area. In a manner similar to that used for the UZ, the 1,300 m of SZ was discretized into 10 mixing cells. The rate of mass flux in the SZ was set equal to the infiltration times the cross-sectional area, which was assumed to be the same as the ore body. The concentration out of the SZ mixing cell at the distance of interest was monitored and stored in the Repository Integration Program output file. Radionuclide concentrations were captured at 150, 600, and 1,300 m downgradient from the Nopal I ore body.

C.3.4 RESULTS

C.3.4.1 Simulations

The Peña Blanca performance-assessment-model simulations were conducted for the scenarios previously described. The model captured uranium and technetium concentrations at various distances from the mine. The data tracking number (DTN) for the results presented in Figures C-6 through C-12 is MO0009MWDPEN01.009 [153039]. The only calibration data consisted of concentrations of uranium reported for water samples collected at the Nopal I mining camp water supply well, as reported in Pickett and Murphy (1999 [110009]). Figure C-6 shows the results of the base-case simulations using the best estimates of site parameters and capture of the radionuclide concentrations at 1,300 m from the ore deposit, approximately the distance of the mining-camp water-supply well. The data show that the total concentration is dominated by ^{238}U, with lesser contributions from other uranium species. The concentration of ^{231}Pa peaks and falls off because of the limited inventory of this radionuclide, Section C.3.3.1, its short half life of 3.25×10^4 years, and the fact that RIP does not include a generation function for ^{231}Pa. The data from Pearcy et al. (1993 [151774]) and Pearcy (1994 [149523]) indicate that, at least laterally, the majority of the uranium released from the ore body has been reprecipitated on fractures in the welded tuff around the ore body. However, there are no rock-analysis data from beneath the ore body, and the fate of uranium in downward percolating water, such as modeled by the Repository Integration Program, cannot be corroborated by field data at this time.

Figures C-7 and C-8 show the concentrations captured at a distance of 150 m and 600 m from the ore body, respectively. These results also show that the relatively short half-life of ^{99}Tc

(2.13×10^5 yr.) means that ^{99}Tc rapidly decays producing only very low, and comparable, concentrations for the majority of the simulation time at the capture distances used in the Peña Blanca model. The data for these flowpaths indicate that uranium is still the dominant species in the water and that the peak concentrations of ^{99}Tc occurred within the first 50,000 years of the simulation. As reflected in Table C-2, the initial peak concentration assumes that there was an accumulated inventory in the ore body at the time it was first exposed to weathering and other degradation processes. Thus, there is an initial peak concentration in the model output because of flushing out the accumulated inventory, shown by a different symbol on Figures C-6 to C-8. However, this peak concentration quickly reduces to negligible at all distances because of the relatively short half-life of ^{99}Tc.

After the initial peak concentration, the concentration of ^{99}Tc is maintained by taking into account ^{99}Tc production from spontaneous fission of ^{238}U (Curtis et al. 1999 [110987]). Because of the long half-life of ^{238}U, the resulting ^{99}Tc concentrations did not noticeably increase or decrease for the remainder of the simulation period but remained at a relatively low concentration throughout the entire 1-million-year simulation period at all capture distances.

The results of the model at the 1,300-m distance, the only distance for which there are corroborating data, were investigated using sensitivity analysis for parameters of interest. Figure C-9 shows the sensitivity of the results to the sorption coefficient (K_d) of material in the UZ. In this simulation, a relatively large K_d value was assumed for uranium (1 m^3/kg), but the K_d for ^{99}Tc was assumed to be zero. The results indicate that the ^{238}U concentrations were significantly reduced, whereas the ^{99}Tc concentrations remained at constant low concentration.

Figure C-10 indicates that reducing the infiltration to 10 percent of the base-case value reduces the amount of uranium released by an order of magnitude but does not significantly affect the ^{99}Tc concentration. Figure C-11 shows that a reduction in uranium solubility by one order of magnitude reduces the uranium concentration, but the release of ^{99}Tc remains at similar levels to that shown for the base case (see Figure C-6). Alternatively, Figure C-12 shows that when the solubility is decreased by reducing the surface area of the inventory of leachable mineral species, both uranium and ^{99}Tc concentrations are reduced.

C.3.4.2 Comparison to the Yucca Mountain Performance Assessment Modeling

In comparing the Repository Integration Program analysis of the Nopal I uranium mine in the Peña Blanca mining district to the Repository Integration Program analysis of Yucca Mountain, as included in the TSPA-VA (DOE 1998 [100550], Volume 3), the following contrasts and similarities are noted:

- The potential Yucca Mountain nuclear-waste repository would contain 154 times more uranium than the Nopal I mine (Murphy and Codell 1999 [149529]).
- The time scale at Nopal I is on the order of 3 million years, as opposed to a prediction time of 1 million years for Yucca Mountain.
- The ^{99}Tc is assumed to be primarily produced naturally from uranium at Nopal I and primarily deposited as a waste inventory of reactor products at Yucca Mountain.

- The hydrogeologic configuration of the Nopal I mine is relatively simple. The ore body is exposed at land surface with an approximately 200 meter-thick UZ above the water table. The SZ at Yucca Mountain is mainly in volcanic rocks at comparable distance from the potential repository; whereas, at Nopal I the SZ is primarily in the Cretaceous limestone found in and beneath the Sierra Peña Blanca.

- There are no naturally radioactive deposits in the host rocks for the potential Yucca Mountain repository. Thus, radionuclide transport calculations through the tuffs below the potential repository horizon are, of necessity, reasoned approximations of what could occur in the event that waste was emplaced at the potential repository. At Nopal I, there is natural uranium that not only dissolves and migrates but also has the potential for the production and transport of conservative daughter products.

- The regional, surface-water-discharge location for the Nopal I ore deposit is approximately 10 km from the deposit versus an approximate 60-km travel distance at Yucca Mountain.

C.3.5 CONCLUSIONS

The Nopal I uranium-ore deposit at the Sierra Peña Blanca was modeled as a natural analogue of the potential Yucca Mountain nuclear-waste repository. The objective of the study was to estimate the potential for migration of uranium species and potential analyzable quantities of dissolved ^{99}Tc using the same modeling techniques used in performance-assessment modeling for the YMP.

The analysis indicates that picogram quantities of ^{99}Tc generated by spontaneous fission of ^{238}U may be detectable at 150, 600, and 1,300 m from the Nopal I ore deposit. In addition, uranium appears to be transported in limited quantities. However, released uranium is apparently exchanging with uranium minerals that precipitate in fractures around the ore deposit. Despite this precipitation, there is sufficient uranium available to be transported through the Cretaceous limestone of north-central Chihuahua State. The analysis indicates that a groundwater-sampling program could provide data with which to estimate realistic transport parameters for the Peña Blanca site. By analogy, these parameters may be a useful tool in estimating the performance assessment of the Yucca Mountain site.

The results at 1,300-m downgradient from the ore body show potential considerable uptake of uranium because of the K_d values used in the model. Given the potential uranium uptake, the results indicate that the estimated concentration of ^{99}Tc may be increased relative to the uranium concentration. However, the model results indicate that the ^{99}Tc concentration would still be a very low, but analyzable concentration (1.E-8 mg/L).

The model results indicate that uranium concentration at an observation point varies directly with the quantity of infiltration and the solubility of the ore body, whether or not the solubility increase is due either to the value for solubility used in the model or to an increase in surface area available for dissolution.

The low concentration of uranium at the mining-camp water-supply well is due either to uranium uptake in the vicinity of the ore deposit or to the location of the well relative to the regional flow direction. Because of the paucity of well data and other hydrogeologic data for the area, the estimated direction and gradient of groundwater flow is highly uncertain. The tentative conclusions developed as a result of the modeling previously described could be enhanced or modified with the implementation of the drilling program described in Section C.2.6. These recommendations were designed to help provide data with which to more accurately define the magnitude and direction of the groundwater gradient in the vicinity of the Nopal I mine. The proposed additional monitor wells are also needed to provide water-sampling locations that will be used to calibrate future performance-assessment modeling in the Peña Blanca area.

C.3.6 FUTURE FIELD INVESTIGATIONS

The results of the modeling indicate that a modest drilling-and-coring program at the Nopal I uranium mine could provide valuable data for use in calibrating the model used in this study to predict the flow and transport of radionuclides released from the Nopal I ore body. Because the Peña Blanca performance-assessment model uses the same methodology as used for Yucca Mountain performance-assessment modeling, calibration of the Peña Blanca model would provide considerable improvement in the confidence in YMP performance-assessment modeling efforts.

Figure C-13 shows a proposed borehole layout for a drilling-and-coring program designed to obtain field data to corroborate this Peña Blanca modeling analysis. The goal of the drilling program would be both to obtain core samples of the ore deposit and the rocks immediately below the ore deposit and to provide locations for obtaining water samples from the Cretaceous limestone, which is the host horizon for the regional groundwater flowing through this region. The key elements of the proposal are as follows:

- Drill a borehole completely through the Nopal I ore deposit and continue to the water table with continuous core-sample recovery.
- Drill boreholes at selected distances east and west of the ore body along the projected gradient of groundwater flow.
- Drill one borehole in an upgradient location to obtain background water-quality data to compare to that obtained in the vicinity of the ore deposit.
- Use a downhole pump to collect water samples for laboratory analysis from the corehole and from all drilled boreholes.
- Rehabilitate the mining camp water supply well to allow collection of water samples for laboratory analysis.
- Collect water samples from the Papalote ranch supply well to obtain laboratory analyses from a location midway between the ore deposit and the probable regional discharge location, the Playa Cuervo.

- Analyze all water samples for the common suite of ionic constituents so that a geochemical balance can be obtained. In addition, analyze the water samples for fluoride (a natural tracer), uranium isotopes, ^{99}Tc, and thorium isotopes.

- Use groundwater-speciation models to analyze the geochemistry and distribution of radioactive isotopes in the flowing groundwater beneath and in the vicinity of the Nopal I ore deposit.

C.4 GENERAL CONCLUSIONS REGARDING THE USE OF NATURAL ANALOGUES IN SUPPORT OF PERFORMANCE ASSESSMENT AT YUCCA MOUNTAIN

The performance-assessment-related investigations described in this Appendix complement the modeling and experimental studies, and literature review and analysis in *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]), *Natural Analogues for the Unsaturated Zone* (CRWMS 2000 [141407]), and Murrell et al. (1999 [130408]). Use of the ongoing research of natural analogues in future confirmation analyses and performance-assessment calculations will provide improved confidence in the potential Yucca Mountain repository concept. Specifically, as discussed in *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]), continued research in the following areas will continue to improve overall confidence:

- Uranium mobility in natural environments, especially in and near analogous ore deposits

- Performance of borosilicate glass in adverse environments

- Metallic-corrosion analysis, especially with respect to alloys specified in the potential repository design

- Mobility of transuranics such as those in natural and anthropogenic reactors

- Fluid-transport parameters in analogous saturated and unsaturated media

- Mobility of analogous colloids

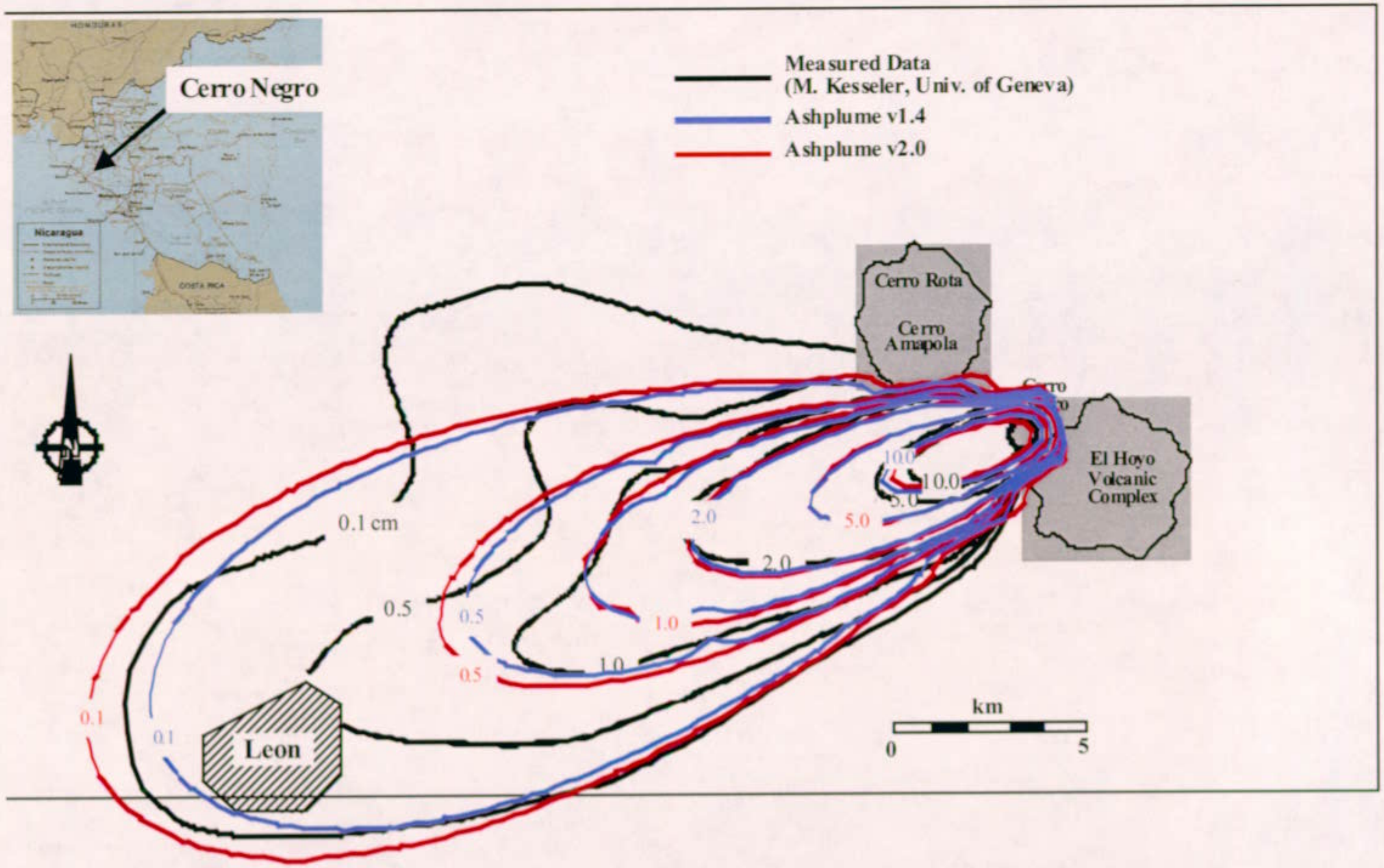
- Natural-retardation processes affecting mobile radionuclides expected to be released from the potential Yucca Mountain repository

- The role of fractures in enhancing or retarding flow and transport of radionuclides in environments analogous to the potential Yucca Mountain repository.

In addition to the investigations described in *Yucca Mountain Site Description* (CRWMS M&O 2000 [137917]) and *Natural Analogues for the Unsaturated Zone* (CRWMS M&O 2000 [141407]), this appendix has addressed specific performance-related analogues that were addressed using performance-assessment models. The location of Yucca Mountain is rather unique hydrogeologically and geographically. The most promising natural analogues are in those related to geologic processes. In particular, the investigations at the Cerro Negro volcano in Nicaragua and at the Nopal I uranium deposit at Peña Blanca near Chihuahua, Mexico. Using the field data from Cerro Negro, the performance-assessment group of the YMP developed a reasonable model representation of a 1995 volcanic event. This model is available to estimate the pattern of ash from a possible volcano at Yucca Mountain and can be simulated for credible scenarios using correlative parameters. Using the model, possible future volcanic events could be modeled as spreading ash downwind from Yucca Mountain, and the consequences of such an event could be analyzed to determine the potential effect on the repository and the surrounding environment.

Radionuclide transport by groundwater is the most likely offsite transport pathway that could possibly affect the performance of the potential Yucca Mountain repository. The Peña Blanca natural-analogue site offers a unique opportunity to examine the potential for groundwater flow and transport of uranium and some of its daughter products in a climatic and geologic setting very similar to that of Yucca Mountain. Both sites are set in volcanic tuff in an oxidizing UZ, and they are in similar desert environments. One major difference between the two sites is that the Yucca Mountain site has had considerably more site characterization of the geologic conditions beneath the site than has the Nopal I site. More detailed site characterization will be needed to more fully establish the utility of this site in comparison to Yucca Mountain.

The Nopal I mine is primarily composed of uraninite, which is essentially the same material as nuclear fuel, and the deposit can be analyzed using the metal-fuel dissolution model used in Yucca Mountain performance assessment. The performance-assessment department conducted scoping calculations for the Peña Blanca site using the same performance-assessment numerical model that was used to evaluate the performance of the potential Yucca Mountain repository. The model attempted to predict the transport of ^{99}Tc, an expected conservative ion that will be liberated from the waste packages to be placed in Yucca Mountain. The results of the model indicate that both ^{99}Tc and some forms of uranium may be detected in groundwater close to the Nopal I mine. Therefore, verification of Peña Blanca as a useful natural analogue will depend on the implementation of a field campaign to collect rock and water samples to corroborate the model results and for a more complete comparison to possible future conditions at Yucca Mountain. These recommended studies will complement those mineralogic and geochemical investigations recommended in *Natural Analogues for the Unsaturated Zone* (CRWMS M&O 2000 [141407]).



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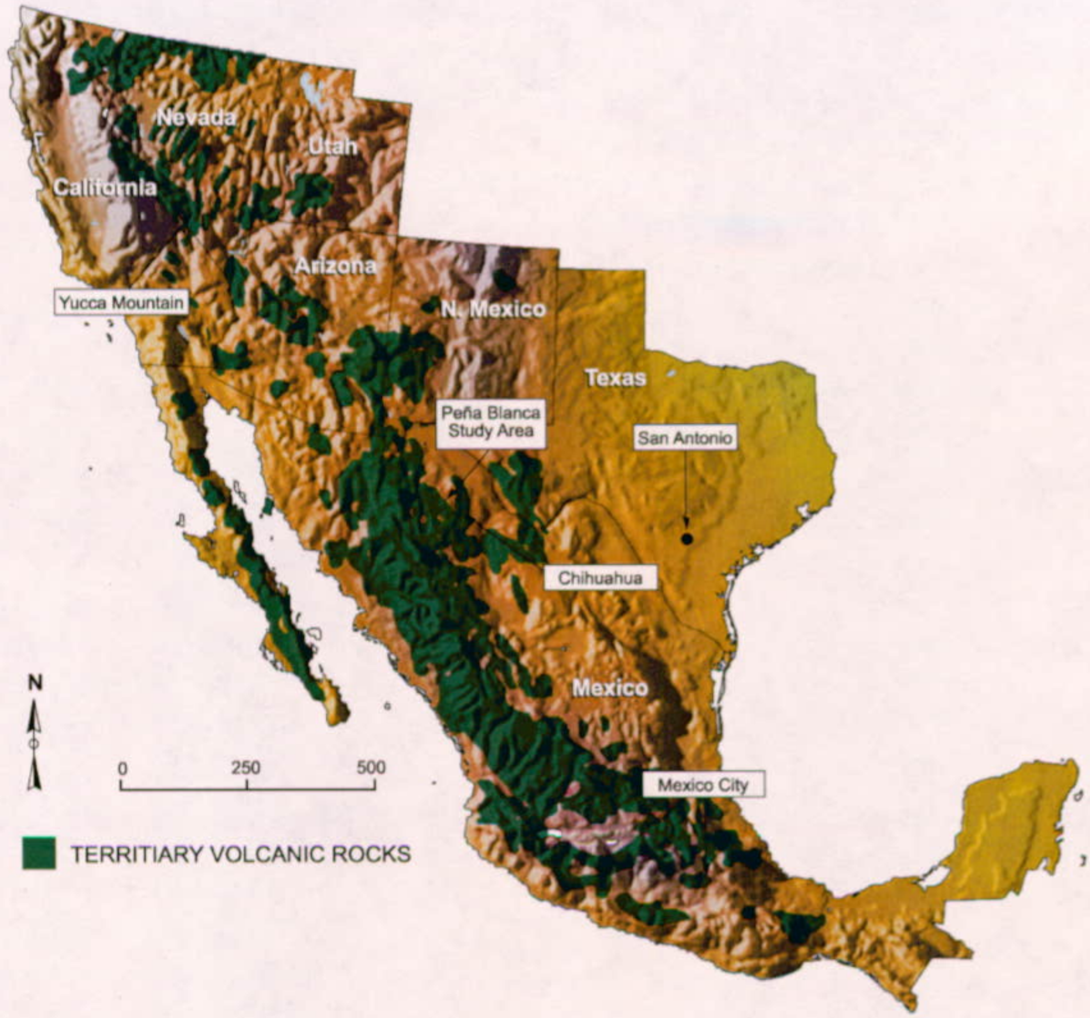
NOTE: Legibility of index map in upper left doesn't detract from the results shown in figure.

Figure C-1. Comparison of the Measured and Calculated Ash-Deposit Thickness for the 1995 Cerro Negro, Nicaragua, Eruption



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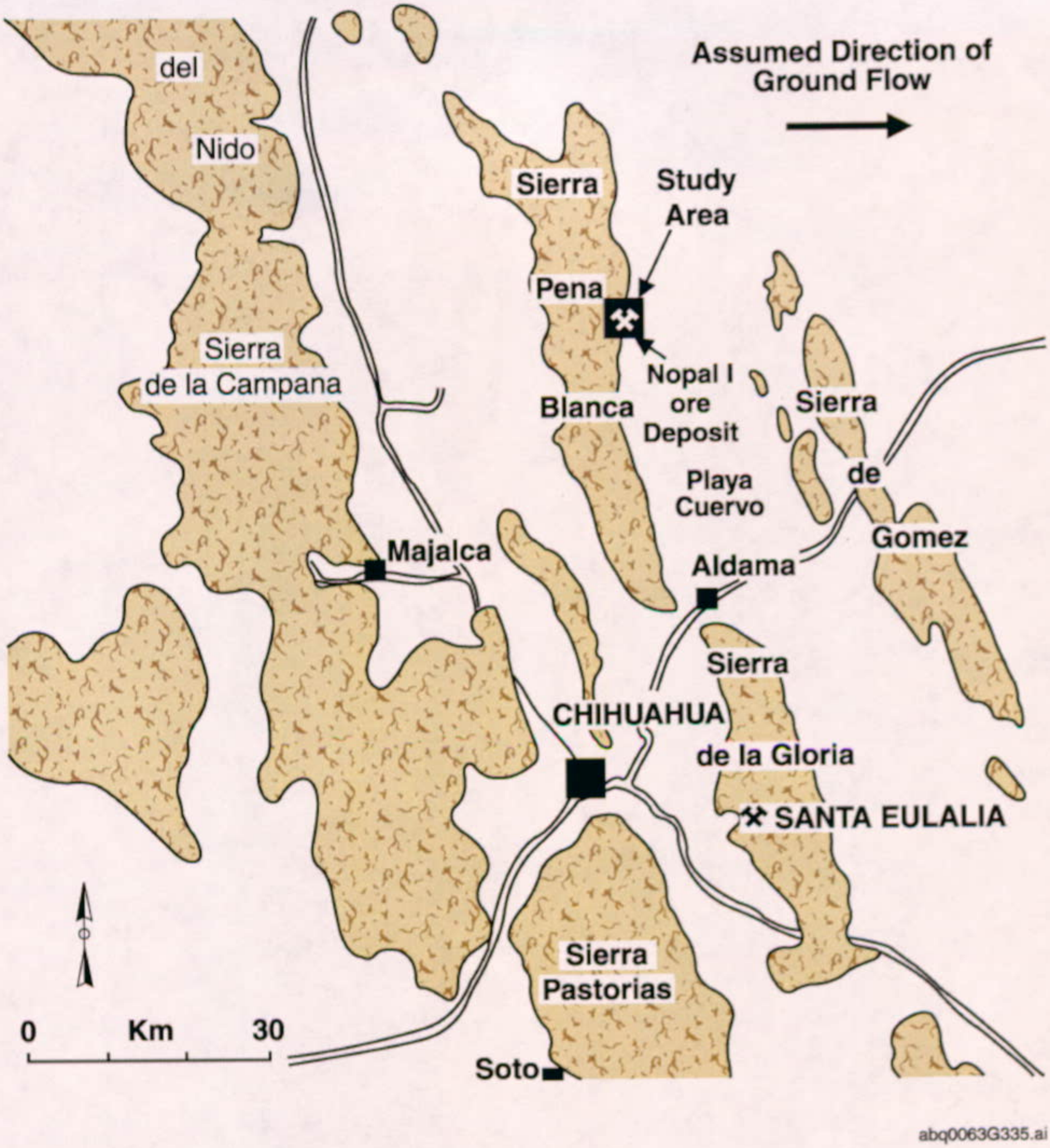
Source: Pearcy 1994 [149523], p.1-2

Figure C-2. Location of the Sierra Peña Blanca in Northern Mexico



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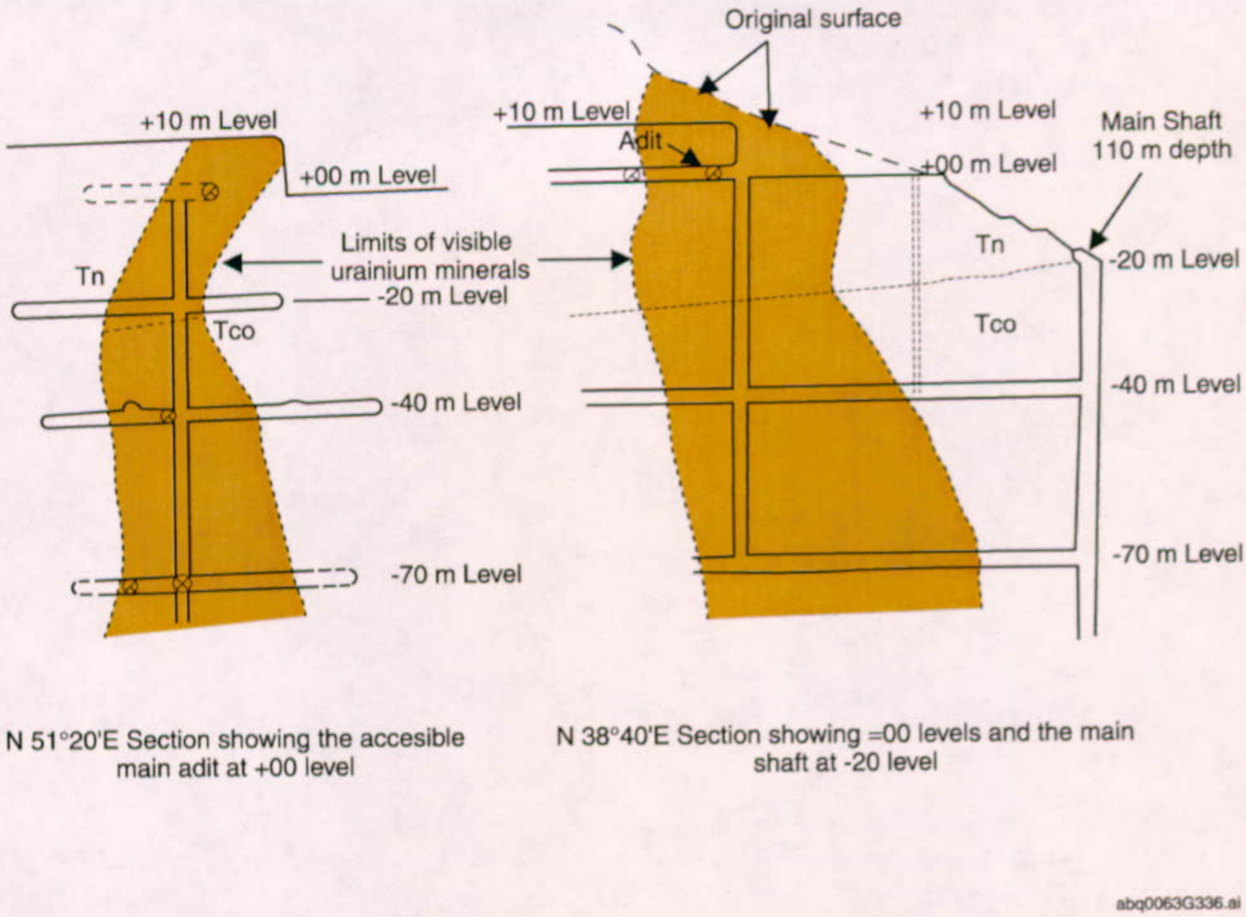
Source: Modified from George-Aniel et al. 1991 [105636]

Figure C-3. Location of the Nopal I Ore Deposit Relative to Chihuahua City



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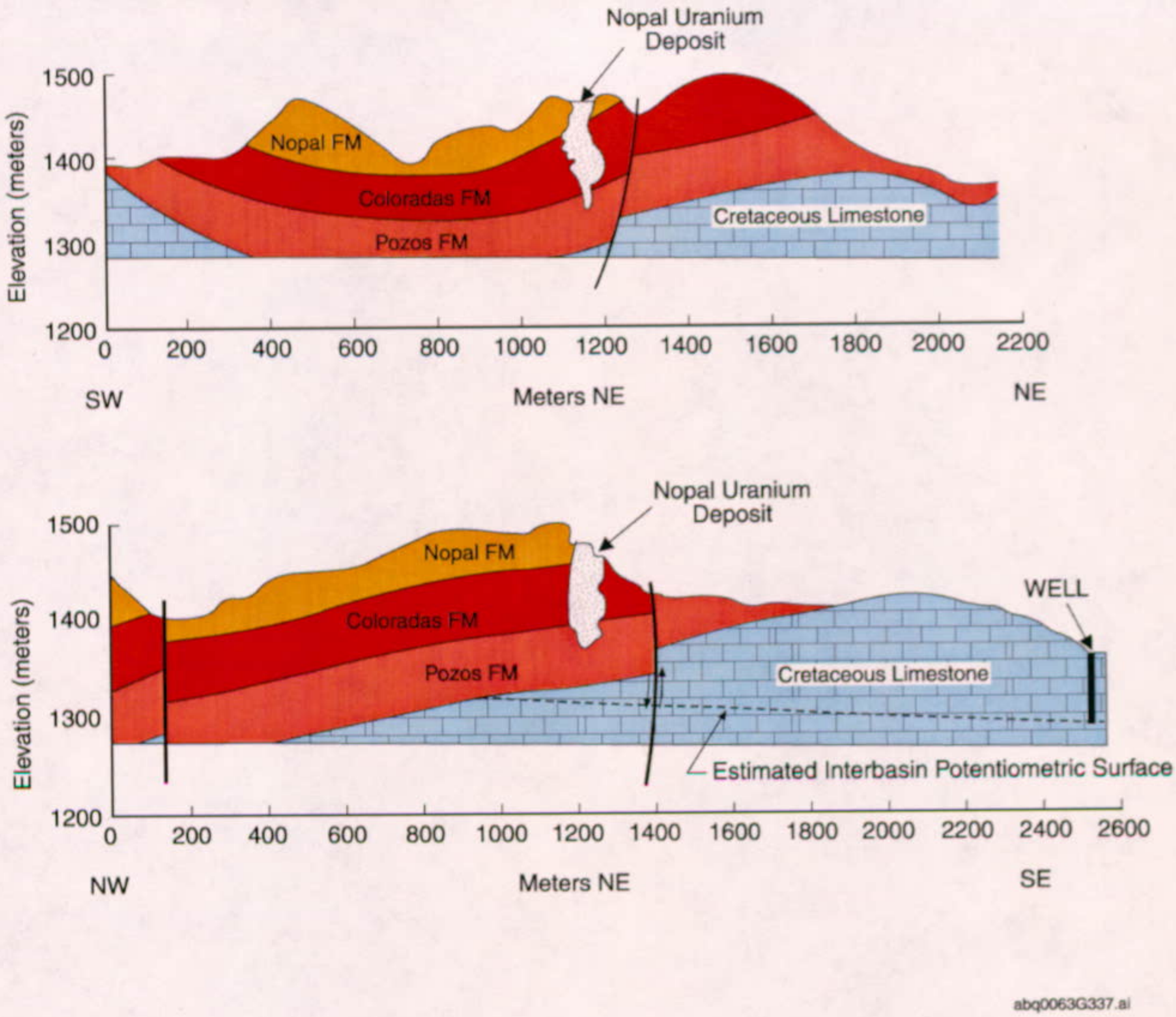
Source: Reyes-Cortez 1997 [149533], p. 205

Figure C-4. Cross-Sectional Views of the Nopal I Uranium Mine Showing the Shafts and Adits



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Source: Pearcy et al. 1993 [151774], p. 1-4

Figure C-5. Hydrogeology of the Nopal I Ore Deposit at Peña Blanca

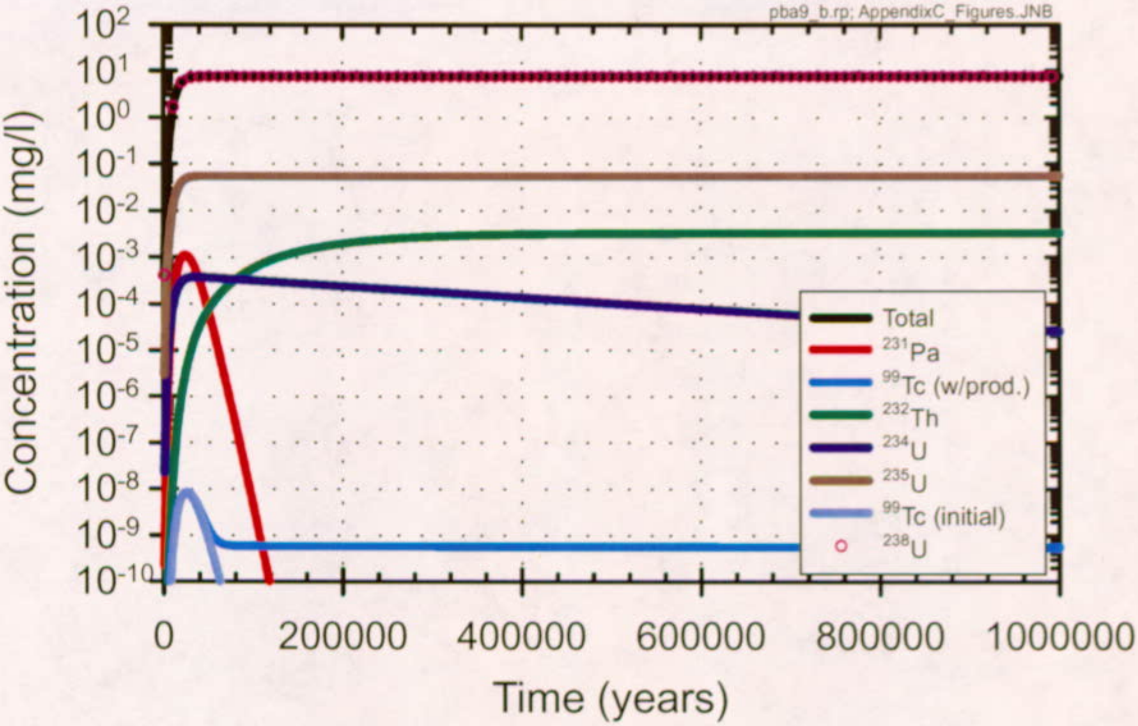

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Figure C-6. Radionuclide Concentrations 1,300 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production

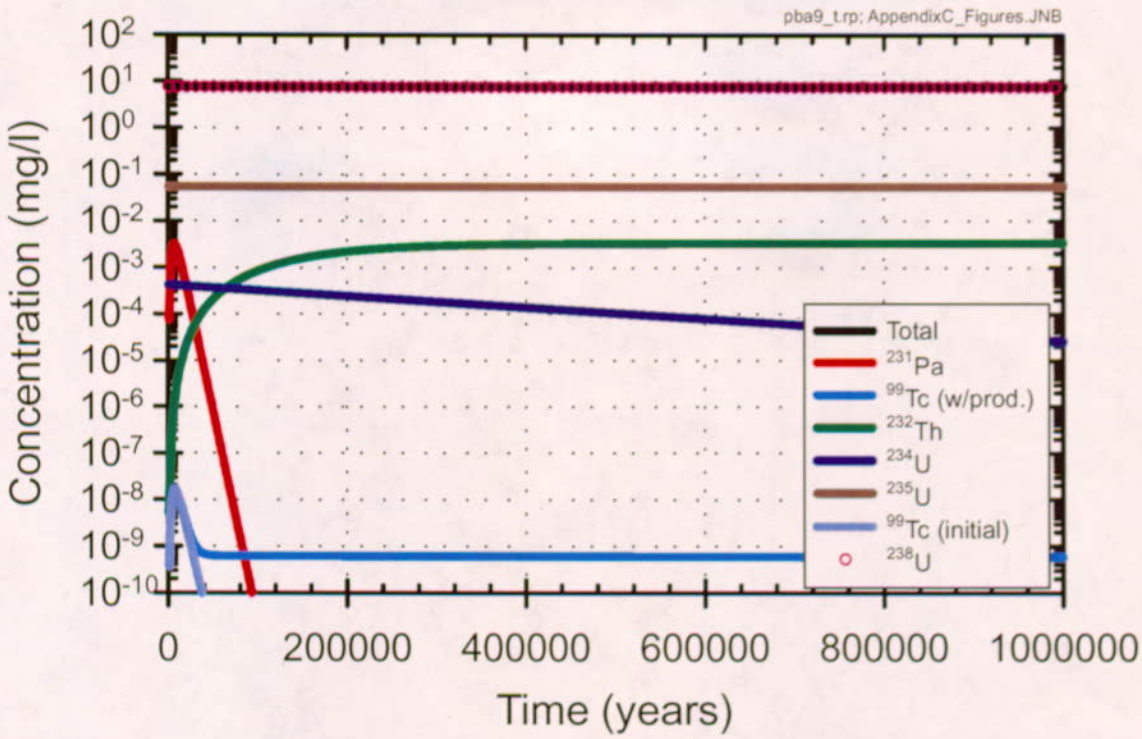

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Figure C-7. Radionuclide Concentrations 600 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production

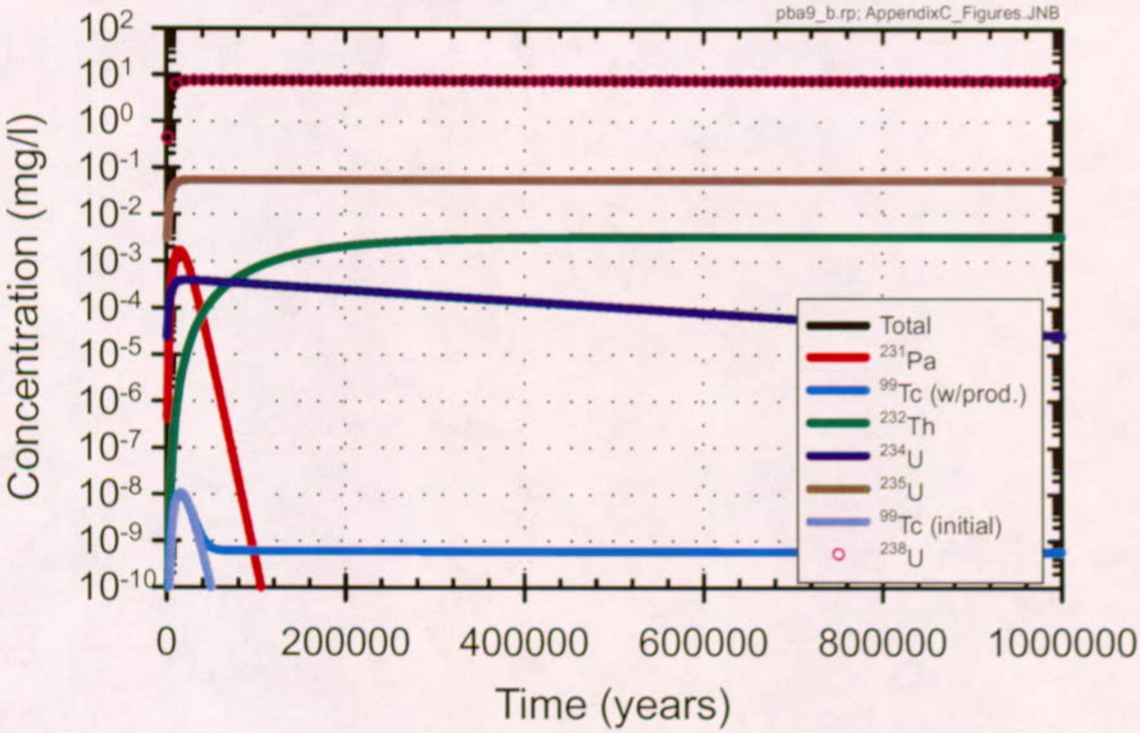

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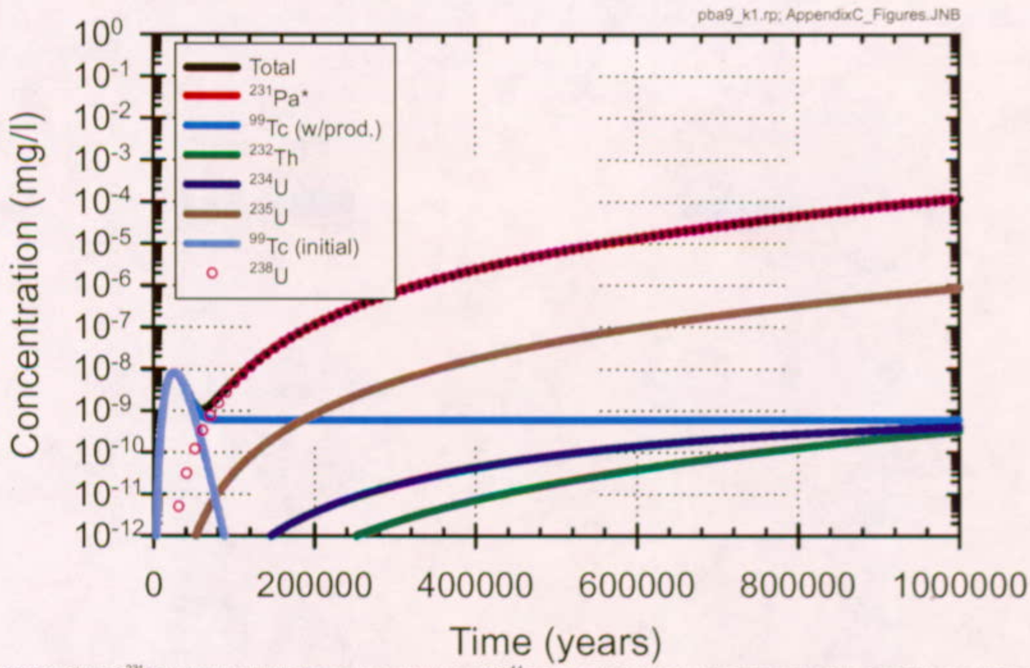
Figure C-8. Radionuclide Concentrations 150 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production



* The concentration of ^{231}Pa was calculated. The calculated values (<10^{-14} mg/l) are below the y-axis minimum and do not appear on this plot.

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Figure C-9. Radionuclide Concentrations 1,300 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production and Showing Sensitivity to K_d

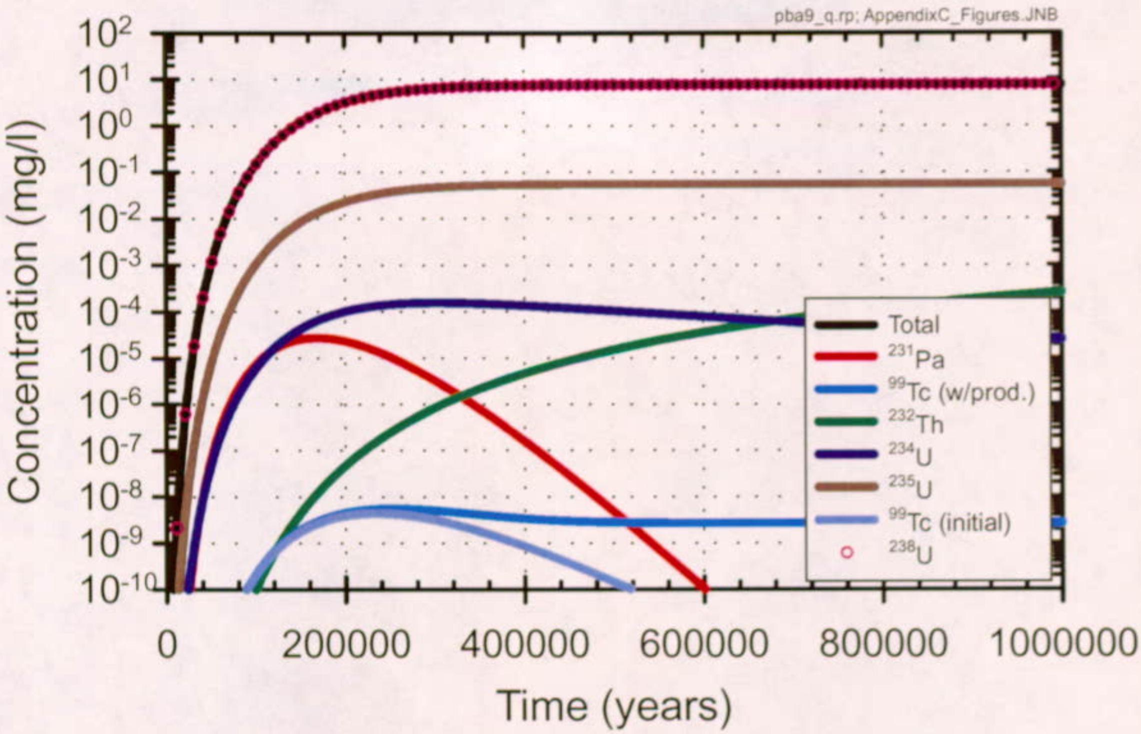

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Figure C-10. Radionuclide Concentrations 1,300 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production and Showing Sensitivity to Infiltration

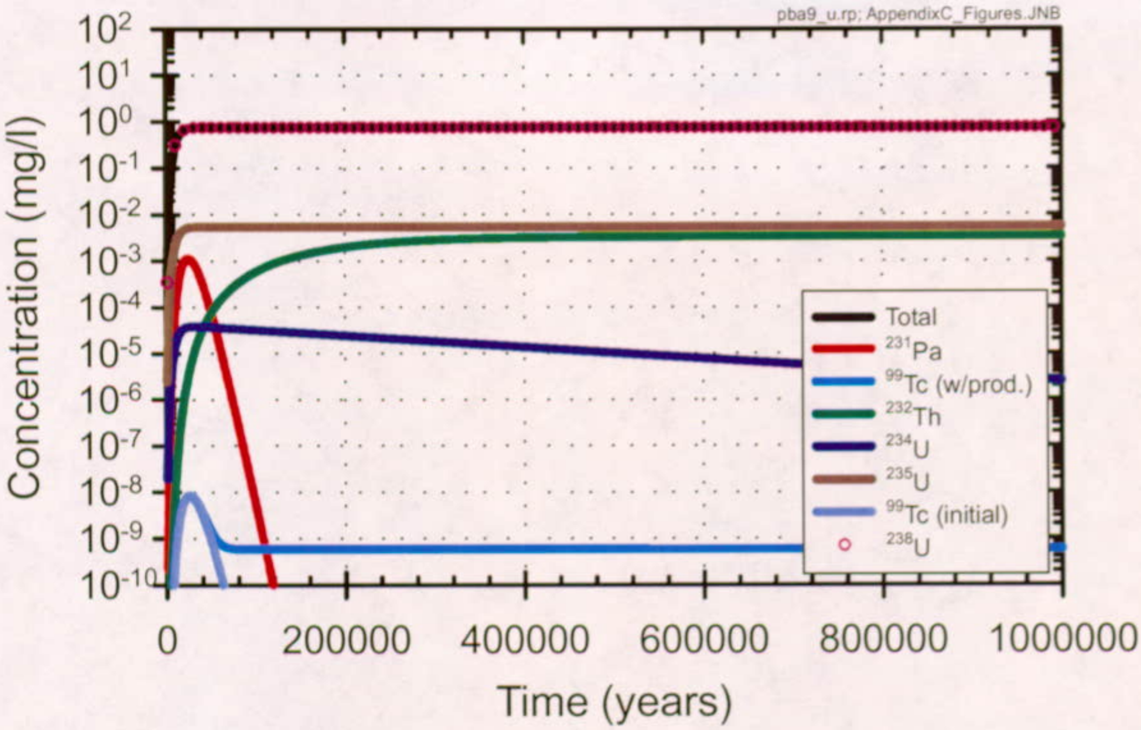

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Figure C-11. Radionuclide Concentrations 1,300 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production and Showing Sensitivity to Uranium Solubility

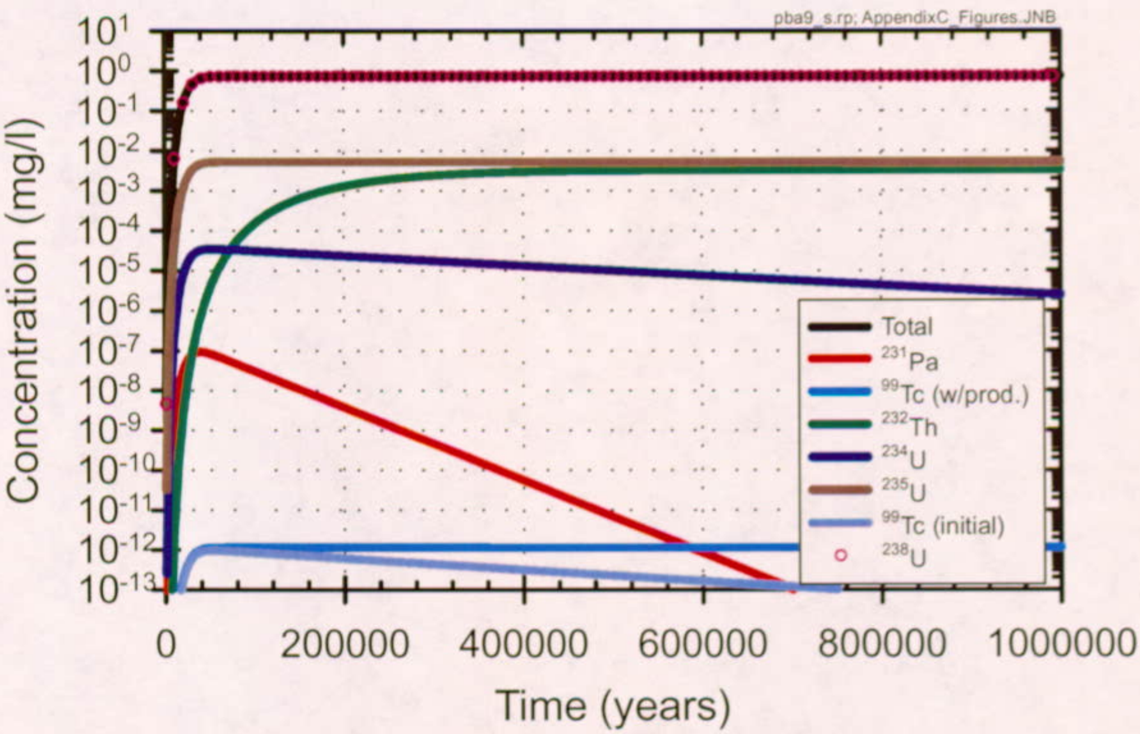

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Figure C-12. Radionuclide Concentrations 1,300 m Downgradient from the Nopal I Ore Deposit Using Constant ^{99}Tc and ^{238}U Production and Showing Sensitivity to Surface Area of the Source Term

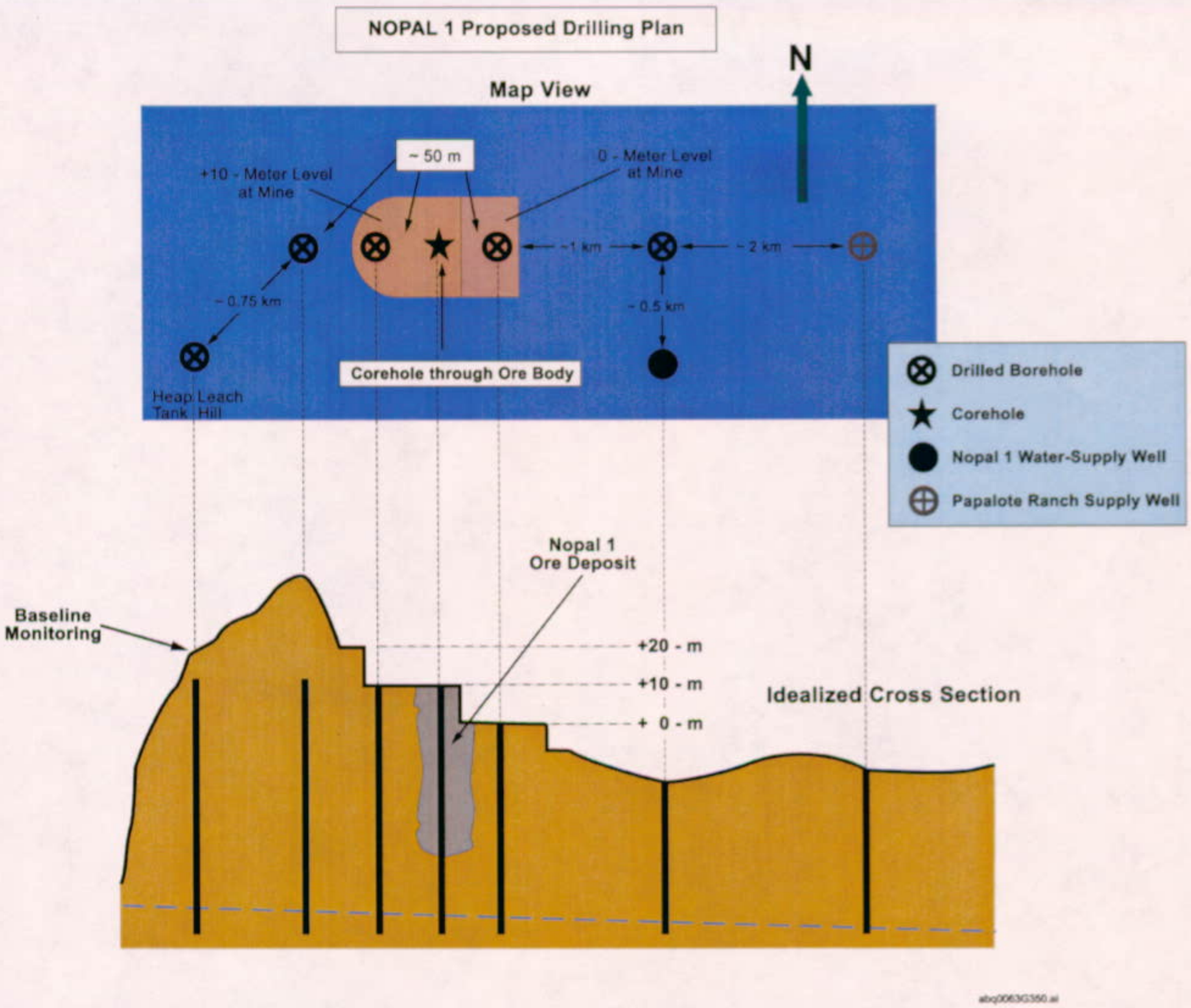

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Figure C-13. Proposed Drilling Plan for Further Study at Nopal I

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APPENDIX D

ISSUE RESOLUTION STATUS REPORTS TRACKING DATABASE

APPENDIX D

ACRONYMS AND ABBREVIATIONS

AMR	Analysis Model Report
BDCF	biosphere dose conversion factor
BIO	biosphere
CLST	container life and source term
DE	disruptive event
DOE	U.S. Department of Energy
EBS	engineered barrier system
ENFE	Evolution of the Near-Field Environment
FEP	feature, event, and process
HLW	high-level radioactive waste
IA	Igneous Activity
IRSR	Issue Resolution Status Report
ISIs	integrated subissues
KTI	key technical issue
NFE	near-field environment
NRC	U.S. Nuclear Regulatory Commission
PMR	Process Model Report
QARD	Quality Assurance Requirements and Description
RDTME	Repository Design and Thermal Mechanical Effects
RSS4	Repository Safety Strategy 4
RT	Radionuclide Transport
SDS	Structural Deformation and Seismicity
SNF	spent nuclear fuel
SZ	saturated zone
SZFT	saturated zone flow and transport
TEF	Thermal Effects on Flow
TSPA	total system performance assessment
TSPA&I	Total System Performance Assessment and Integration
TSPA-SR	Total System Performance Assessment-Site Recommendation

APPENDIX D

ACRONYMS AND ABBREVIATIONS (Continued)

USFUIC	UZ and SZ Flow Under Isothermal Conditions
UZ	unsaturated zone
UZFT	unsaturated zone flow and transport
WFD	waste form degradation
WPD	waste package degradation

APPENDIX D

ISSUE RESOLUTION STATUS REPORTS TRACKING DATABASE

The Issue Resolution Status Reports (IRSR) Tracking Database is designed to track how U.S. Nuclear Regulatory Commission (NRC) subissues within the Key Technical Issues (KTIs) have been addressed by total system performance assessment (TSPA).

D.1 ELEMENTS OF ISSUE RESOLUTION STATUS REPORTS TRACKING DATABASE

D.1.1 ISSUE RESOLUTION STATUS REPORTS

The NRC strategic plan calls for the early identification and resolution, at the staff level, of issues before the receipt of a potential license application to construct a geologic repository. The NRC's high-level radioactive waste (HLW) program has focused prelicensing work on those topics most critical to the postclosure performance of the potential geologic repository; these topics are called KTIs. An important step in the staff's approach to issue resolution is to provide the U.S. Department of Energy (DOE) with feedback regarding issue resolution before the forthcoming Site Recommendation and License Application. IRSRs are the primary mechanism that the NRC staff will use to provide DOE with feedback on KTI subissues. IRSRs focus on issue resolution and the status of resolution, including areas of agreement or when the staff has comments or questions.

D.1.1.1 Key Technical Issues and Subissues

The NRC has identified 10 KTIs. Nine of these issues comprise technical questions; the tenth is a nontechnical issue related to development of the U.S. Environmental Protection Agency standard. Each KTI is subdivided into a number of subissues (Table D.1-1). The KTIs are:

- Container Life and Source Term (CLST)
- Evolution of the Near-Field Environment (ENFE)
- Igneous Activity (IA)
- Radionuclide Transport (RT)
- Repository Design and Thermal Mechanical Effects (RDTME)
- Structural Deformation and Seismicity (SDS)
- Thermal Effects on Flow (TEF)
- Total System Performance Assessment and Integration (TSPA&I)
- UZ and SZ Flow Under Isothermal Conditions (USFUIC)
- Activities Related to Development of the U.S. Environment Protection Agency Standard.

Table D.1-1. Subissues in U.S. Nuclear Regulatory Commission Key Technical Issues

KTI	Subissue Identifier	Subissue
Unsaturated and Saturated Flow under Isothermal Conditions (USFIC)	USFIC1	Climate change
	USFIC2	Hydrologic effects of climate change
	USFIC3	Present-day shallow infiltration
	USFIC4	Deep percolation (present and future)
	USFIC5	Saturated zone ambient flow conditions and dilution processes
	USFIC6	Matrix diffusion
Thermal Effects on Flow (TEF)	TEF1	Sufficiency of thermal-hydrologic testing program to assess thermal reflux in the near field
	TEF2	Sufficiency of thermal-hydrologic modeling to predict the nature and bounds of thermal effects on flow in the near field
	TEF3	Adequacy of TSPA with respect to thermal effects on flow
Evolution of the Near-Field Environment (ENFE)	ENFE1	Effects of coupled thermal-hydrologic-chemical processes on seepage and flow
	ENFE2	Effects of coupled thermal-hydrologic-chemical processes on WP chemical environment
	ENFE3	Effects of coupled thermal-hydrologic-chemical processes on chemical environment for radionuclide release
	ENFE4	Effects of thermal-hydrologic-chemical processes on radionuclide transport (RT) through engineered and natural barriers
	ENFE5	Coupled thermal-hydrologic-chemical processes affecting potential nuclear criticality in the near field
Container Life and Source Term (CLST)	CLST1	The effects of corrosion processes on the lifetime of the containers
	CLST2	The effects of phase instability and initial defects on the mechanical failure and lifetime of the containers
	CLST3	The rate at which radionuclides in spent nuclear fuel are released from the engineered barrier subsystem through the oxidation and dissolution of spent fuel
	CLST4	The rate at which radionuclides in high-level waste (HLW) glass are released from the engineered barrier subsystem
	CLST5	The effects of in-package criticality on waste package and engineered barrier subsystem performance
	CLST6	The effects of alternate engineered barrier subsystem design features on container lifetime and radionuclide release from the engineered barrier subsystem
Radionuclide Transport (RT)	RT1	RT through porous rock
	RT2	RT through alluvium
	RT3	RT through fractured rock
	RT4	Nuclear criticality in the far field
Total System Performance Assessment and Integration (TSPAI)	TSPAI1	System description and demonstration of multiple barrier
	TSPAI2	Scenario analysis within the TSPA methodology
	TSPAI3	Model abstraction within the TSPA methodology
	TSPAI4	Demonstration of the overall performance objective
Igneous Activity (IA)	IA1	Probability of future igneous activity
	IA2	Consequences of igneous activity within the repository setting

Table D.1-1. Subissues in U.S. Nuclear Regulatory Commission Key Technical Issues (Continued)

KTI	Subissue Identifier	Subissue
Structural Deformation and Seismicity (SDS)	SDS1	Faulting
	SDS2	Seismicity
	SDS3	Fracturing and structural framework of the geologic setting
	SDS4	Tectonic framework of the geologic setting
Repository Design and Thermal-Mechanical Effects (RDTME)	RDTME1	Implementation of an effective design control process within the overall quality assurance program
	RDTME2	Design of the geologic repository operations area for the effects of seismic events and direct fault disruption
	RDTME3	Thermal-mechanical effects on underground facility design and performance
	RDTME4	Design and long-term contribution of repository seals in meeting post-closure performance objectives

Source: NRC 2000 [149372], Appendix B

D.1.1.2 Subsystems and Integrated Subissues

As stated in the TSPA&I IRSR Rev 2 (NRC 2000 [149372]), the NRC has been developing a systematic approach to reviewing DOE's TSPAs. The approach currently undertaken by the NRC staff is hierarchical, as illustrated in TSPA&I IRSR, Rev 2 (NRC 2000 [149372], Figure 3, p. 146). As shown in said figure, there are three repository subsystems: engineered system, geosphere, and biosphere. Each of these subsystems is further subdivided into discrete components of the respective subsystems: engineered barriers that make up the engineered system; unsaturated zone flow and transport, saturated zone flow and transport, and direct release to the geosphere; and the dose calculation for the biosphere. At the base of the hierarchy are the 14 integrated subissues (ISIs) of the repository system that need to be appropriately abstracted into a TSPA. The relationship between these ISIs and the NRC KTIs is illustrated in Table D.1-2.

D.1.2 TOTAL SYSTEM PERFORMANCE ASSESSMENT

D.1.2.1 Process Model Reports and Analysis Model Report

Nine Process Model Reports (PMRs) have been developed to summarize the technical basis for each of the process models supporting the TSPA model that will be used to evaluate the postclosure performance of a potential repository at Yucca Mountain. Details of these individual models are documented in supporting Analysis Model Reports (AMRs). These Process Model Reports cover the following areas:

- *Integrated Site Model Process Model Report* (CRWMS M&O 2000 [146988])
- *Unsaturated Zone Flow and Transport Model Process Model Report* (CRWMS M&O 2000 [145774])
- *Near Field Environment Process Model Report* (CRWMS M&O 2000 [146589])

Table D.1-2. Integrated Subissues Relationships to Key Technical Issues

KTI * Subissue	Integrated Subissue													
	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3
USFIC1					■									■
USFIC2								■						■
USFIC3					■									■
USFIC4			■		■	■	■							
USFIC5								■	■			■		■
USFIC6							■	■	■					
TEF1	■		■		■	■								
TEF2	■		■		■	■								
TEF3														
ENFE1			■		■	■								
ENFE2	■		■											
ENFE3			■	■										
ENFE4				■			■							
ENFE5				■										
CLST1	■	■	■							■				
CLST2	■	■								■				
CLST3			■	■										
CLST4			■	■										
CLST5		■		■										
CLST6	■	■	■	■										
RT1							■		■					
RT2									■					
RT3							■		■					■
RT4							■		■					
TSPAI1														
TSPAI2														

Table D.1-2. Integrated Subissues Relationships to Key Technical Issues (Continued)

TSPAI3														
TSPAI4														
IA1														
IA2														
SDS1														
SDS2														
SDS3														
SDS4														
RDTME1														
RDTME2														
RDTME3														
RDTME4														

ENG1	ENG - Degradation of Engineered Barriers	SZ1	GEO - Flow Paths in the SZ
ENG2	ENG - Mechanical Disruption of Engineered Barriers	SZ2	GEO - RT in the SZ
ENG3	ENG - Quantity and Chemistry of Water Contacting the Waste Packages and Waste Forms	Direct1	GEO - Volcanic Disruption of Waste Packages
ENG4	ENG - Radionuclide Release Rates and Solubility Limits	Direct2	GEO - Airborne Transport of Radionuclides
UZ1	GEO - Spatial and Temporal Distribution of Flow	Dose1	BIO - Dilution of Radionuclides in Groundwater
UZ2	GEO - Flow Paths in the UZ	Dose2	BIO - Redistribution of Radionuclides in Soil
UZ3	GEO - RT in the UZ	Dose3	BIO - Lifestyle of the Critical Group

Source: TSPA&I IRSR REV. 2 (NRC 2000 [149372])

- *Engineered Barrier System Degradation, Flow, and Transport Process Model Report* (CRWMS M&O 2000 [145796])

- *Waste Package Degradation Process Model Report* (CRWMS M&O 2000 [138396])

- *Waste Form Degradation Process Model Report* (CRWMS M&O 2000 [138332])

- *Saturated Zone Flow and Transport Process Model Report* (CRWMS M&O 2000 [145738])

- *Biosphere Process Model Report* (CRWMS M&O 2000 [151615])

- *Disruptive Events Process Model Report* (CRWMS M&O 2000 [141733]).

D.1.2.2 Process Model Factors

A set of 26 process model factors for the nominal scenario was identified as described in the Repository Safety Strategy 4 (RSS4). Table D.1-3 lists the set of process model factors and PMRs.

Table D.1-3. Process Model Factors and Related Key Technical Issues Subissues

PMR	Process Model Factor for the Nominal Scenario	Related KTI Subissues
UZ Flow and Transport Model (CRWMS M&O 2000 [145774]	Climate	USFIC1, USFIC2, TSPAI1, TSPAI2, TSPAI3
	Net Infiltration	USFIC3, TSPAI1, TSPAI2, TSPAI3
	UZ Flow	SDS3, TSPAI1, TSPAI2, TSPAI3
	Coupled Effects on UZ Flow	ENFE1, ENFE4, RT1, RT3, TEF2, TEF3, TSPAI1, TSPAI2, TSPAI3
	Seepage into Emplacement Drifts	USFIC4, TSPAI1, TSPAI2, TSPAI3
	Coupled Effects on Seepage	USFIC6, TSPAI1, TSPAI2, TSPAI3
SZ Flow and Transport (CRWMS M&O 2000 [145738]	SZ Flow	SDS3, USFIC2, USFIC5, TSPAI1, TSPAI2, TSPAI3
	Changes to SZ Flow	TSPAI1, TSPAI2, TSPAI3
	SZ RT	RT2, RT3, RT4, USFIC6, TSPAI1, TSPAI2, TSPAI3
Disruptive Events (CRWMS M&O 2000 [141733])	Seismic Activity	SDS1, SDS2, CLST2, CLST6, TSPAI1, TSPAI2, TSPAI3
	Igneous Activity	IA1, IA2, SDS4, CLST2, CLST6, TSPAI1, TSPAI2, TSPAI3
	Inadvertent Human Intrusion	CLST2, CLST6, TSPAI1, TSPAI2, TSPAI3
Biosphere (CRWMS M&O 2000 [151615]	Biosphere Dose Conversion Factors	IA2, TSPAI1, TSPAI2, TSPAI3

Table D.1-3. Process Model Factors and Related Key Technical Issues Subissues (Continued)

PMR	Process Model Factor for the Nominal Scenario	Related KTI Subissues
Engineered Barrier System (CRWMS M&O 2000 [145796])	In-Drift Physical and Chemical Environment	CLST3, CLST6, ENFE2, ENFE3, ENFE4, RDTME3, TEF1, TEF2, TEF3, TSPAI1, TSPAI2, TSPAI3
	In-Drift Moisture Distribution	CLST1, CLST4, ENFE1, TSPAI1, TSPAI2, TSPAI3
Waste Form (CRWMS M&O 2000 [138332])	Cladding Degradation and Performance	TSPAI1, TSPAI2, TSPAI3
	Colloid-Associated Radionuclide Concentrations	TSPAI1, TSPAI2, TSPAI3
	Commercial Spent Nuclear Fuel Degradation and Performance	CLST3, TSPAI1, TSPAI2, TSPAI3
	Defense High-Level Radioactive Waste Degradation and Performance	CLST4, TSPAI1, TSPAI2, TSPAI3
	Dissolved Radionuclide Concentrations	TSPAI1, TSPAI2, TSPAI3
	DOE-Owned Spent Nuclear Fuel Degradation and Performance	TSPAI1, TSPAI2, TSPAI3
	In-Package Environments	ENFE3, TSPAI1, TSPAI2, TSPAI3
	Radionuclide Inventory and Distribution	TSPAI1, TSPAI2, TSPAI3
Waste Package (CRWMS M&O 2000 [138396])	Waste Package Degradation and Performance	CLST1, CLST2, CLST6, TSPAI1, TSPAI2, TSPAI3
	Drip Shield Degradation and Performance	TSPAI1, TSPAI2, TSPAI3
Near Field Environments (CRWMS M&O 2000 [146589])	Coupled Effects on UZ Flow	ENFE4, RDTME3, TEF1, TSPAI1, TSPAI2, TSPAI3
	Coupled Effects on Seepage	ENFE1, ENFE2, TEF1, TEF2, TSPAI1, TSPAI2, TSPAI3
	Coupled Effects on UZ RT	ENFE4, TEF1, TSPAI1, TSPAI2, TSPAI3

D.1.2.3 Addressing Key Technical Issues Subissues

Guidance for the NRC review of the TSPA is contained in the TSPA&I IRSR (NRC 2000 [149372]). The TSPA&I IRSR describes an acceptable methodology for assessing repository performance and for using these assessments to demonstrate compliance with the overall performance objective and requirements for multiple barriers. The TSPA&I IRSR subissues are integrated into the nine PMRs described in Section D.1.2.1. Table D.1-4 lists all TSPA&I IRSR subissues, lists documentation related to the TSPA&I IRSR acceptance criteria, and provides a brief summary of the approach to resolve the TSPA&I IRSR acceptance criteria.

Each PMR discusses how analyses and calculations address relevant KTI subissues. The detailed discussions are included in Chapter 4, Relationship to NRC Issue Resolution Status Reports, of each PMR. A crosswalk of KTI subissues to PMRs is provided in Table D.1-5. These subissues are also mapped to applicable process model factors as shown in Table D.1-3.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
SUBISSUE 1 - System Description and Demonstration of Multiple Barriers		
Transparency and Traceability of the Analysis		
TSPA Documentation Style, Structure, and Organization		
T1) Documents and reports are complete, clear, and consistent.	TSPA-SR Technical Report, all PMRs, all AMRs	The TSPA-SR Technical Report, PMRs, and AMRs were carefully structured to be complete, clear, and consistent. Section 1 of the TSPA-SR Technical Report explains in detail what total performance assessment is and why it is applicable to repository development. Section 2 describes the specific way in which the general performance assessment approach was adopted for the total system performance assessment for the site recommendation. It also describes the traceability of the information used in the model. Reviews of the TSPA-SR Technical Report, PMRs, and AMRs included checks for completeness, clarity and consistency.
T2) Information is amply cross referenced.	TSPA-SR Technical Report, all PMRs, all AMRs	The TSPA-SR Technical Report, PMRs, and AMRs contain ample references to data sources, codes, assumptions, and conclusions. Extensive use is made of charts and tables to ensure that the total system performance assessment process is easy to visualize. Tables describe the relationship between documents to enhance understanding and traceability.
Features, Events, and Processes Identification and Screening		
T1) The screening process by which FEPs were included or excluded from the TSPA is fully described.	TSPA-SR Technical Report Section 2.1.1, Appendix B	Guidelines were established to ensure that the screening basis and content for each primary FEP was sufficient to satisfy the screening criteria. Section 2.1.1 of the TSPA-SR Technical Report describes the implementation of the features, events, and processes approach including screening of FEPs. It describes how each of the primary FEPs was screened for inclusion or exclusion in the TSPA based on three criteria. The criteria, described in detail in Section B.4 of Appendix B, are regulatory, probability and consequence. The screening results and rationale for the decisions is documented in the FEP AMRs.
T2) Relationships between relevant FEPs are fully described.	TSPA-SR Technical Report Section 2.1.1, FEP AMRs	The TSPA-SR Technical Report includes tables of included FEPs for each TSPA-SR component. If an included FEP affects more than one component it is listed under each relevant component. Also included is a description of the conceptual model based on included FEPs and a description of the TSPA implementation of the conceptual model based on the included FEPs. Section 2.1.1 of the TSPA-SR Technical Report describes the relationship between primary and secondary FEPs. The FEP AMRs provide additional documentation including the TSPA disposition of FEPs, IRSR issues relevant to specific FEPs, and analysis and discussion on specific FEPs.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
Abstraction Methodology		
T1) The levels and method(s) of abstraction are described starting from assumptions defining the scope of the assessment down to assumptions concerning specific processes and the validity of given data.	TSPA-SR Technical Report Section 2 and 3, all PMRs	For each model described in the PMRs, descriptions are provided of process models and, if the models are abstracted, descriptions of the abstractions of the models. The description includes a summary of data and assumptions used to construct models. The AMRs describing the models and the abstracted models provide additional details regarding data and assumptions. Section 2 of the TSPA-SR Technical Report provides a general discussion of the approach used to analyze the potential repository. Section 2.2.1 provides descriptions and figures showing the flow of information from the most basic detailed level to the total system model. Section 3 describes the individual components of the model, including implementation in the TSPA. These sections describe the abstraction models, if applicable, and describe the connection with other TSPA model components.
T2) A mapping (e.g., a road map diagram, a traceability matrix, a cross-reference matrix) is provided to show what conceptual features (e.g., patterns of volcanic events) and processes are represented in the abstracted models, and by what algorithms.	TSPA-SR Technical Report Section 2 and 3	In TSPA-SR Technical Report Section 3, the PMRs and supporting AMRs provide descriptions of the basis for decisions and assumptions that were made during the abstraction process. Section 3 addresses the major features, processes, and conceptualization of the nine process areas and the implementation of these components into the performance assessment analyses. Section 2 describes the different computer codes used to represent the various TSPA component models.
T3) An explicit discussion of uncertainty is provided to identify which issues and factors are of most concern or are key sources of disagreement among experts.	TSPA-SR Technical Report Sections 2.2.1, 3, and 5.1, all PMRs	The TSPA-SR Technical Report and the PMRs provide a discussion of uncertainties and limitations for the major process models included in the report. Section 2.2.1 describes that a key feature of the methodology used is the approach utilized to pass uncertainty at one level to uncertainty at another level. For each component model described in Section 3, the treatment of uncertainty and variability is discussed. Section 5.1, Uncertainty Importance Analysis, presents the results of analyses intended to identify those stochastic variables that have the greatest impact on the output of the TSPA-SR model. When used, expert elicitations were determined to be subject to the quality assurance program as described in the Quality Assurance Requirements Document (QARD). Appendix C of the QARD and implementing procedures for expert elicitation were developed using the guidance provided in NUREG-1563 (Kotra et al. 1996 [100909]).
Data Use and Validity		
T1) The pedigree of data from laboratory tests, natural analogs, and the site is clearly identified.	All PMRs	The PMRs and AMRs that support the TSPA explicitly identify the source and status of data. The PMRs summarizes the quality assurance status of the data and software used in the component models. The AMRs provide additional details regarding the pedigree of data. The qualification status of input data and references are tracked in the electronic Data Input Reference System (DIRS) in accordance with *Managing Technical Product Inputs,* AP-3.15Q [153184]. Data qualification was performed in accordance with procedure *Qualification of Unqualified Data and the Documentation of Rationale for Accepted Data,* AP-SIII.2Q [103748].

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T2) Input parameter development and basis for their selection is described.	All PMRs	The PMRs discuss input parameter development and the basis for using the parameters. The AMRs describing the models provide additional details regarding input parameter development and the basis for input selection.
T3) A thorough description of the method used to identify performance confirmation program parameters.	Performance Confirmation Plan	The *Performance Confirmation Plan* (CRWMS M&O 2000 [150657]) specifically addresses the methodology for identifying and selecting parameters that are important to performance based upon TSPA sensitivity analyses and the repository safety strategy. Methods used to collect information for each parameter will be described by the performance confirmation plan or relevant supporting documents to support the license application. Performance confirmation test selection and rationale is also described in the plan based upon the significance of the parameter being measured, and the ability of the test to distinguish construction, emplacement, or time dependent changes in the parameter significant to performance.
Assessment Results		
T1) PA results (i.e., the peak expected annual dose within the compliance period) can be traced back to applicable analyses that identify the FEPs, assumptions, input parameters, and models in the PA.	TSPA-SR Technical Report Sections 2.2, 3, and 4, All PMRs	The TSPA-SR Technical Report describes features, processes, conceptual models, and their implementation into the TSPA. This discussion will be based in part on information provided by the PMRs. Section 4 provides the results of the performance evaluations and includes a systematic analysis of the contribution of each component of the total system dose. The models that are integrated into the TSPA model are described in detail in Section 3 including a discussion of the FEPs relevant to each model component. Section 2.2 provides a detailed description of the method used to analyze the repository in the TSPA including the approach to combine the individual processes into an overall model and computer code.
T2) The PA results include a presentation of intermediate results that provide insight into the assessment (e.g., results of intermediate calculations of the behavior of individual barriers).	TSPA-SR Technical Report Section 4, All PMRs	Section 4 of the TSPA-SR Technical Report provides performance analysis results for the total system and includes intermediate results for the components of the system.
Code Design and Data Flow		
T1) The flow of information (input and output) between the various modules is clearly described.	TSPA-SR Technical Report Section 2.2	Section 2.2.1 of the TSPA-SR Technical Report provides a description of information flow between component models. Figure 2.2-1 illustrates the information major connections, abstractions and information feeds. Figure 2.2-2 provides a detailed description of information flow in the TSPA-SR, showing the principal pieces of information passed between the various component models.
T2) Supporting documentation (e.g., user's manuals, design documents) clearly describes code structure and relationships between modules.	TSPA-SR Technical Report	The TSPA-SR Technical Report describes the TSPA code and provides references to supporting documentation such as the user's guide. Section 2.2.2, Code Architecture, discusses the coding methods and couplings used for the major components. References are provided in Section 7.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
Demonstration Of Multiple Barriers		
Will be developed in Revision 3 of the IRSR		
SUBISSUE 2 - Scenario Analysis		
Identification of an Initial Set of Processes and Events Data		
1) DOE has identified a comprehensive list of processes and events that: (1) are present or might occur in the Yucca Mountain region and (2) includes those processes and events that have the potential to influence repository performance.	TSPA-SR Technical Report 2.1.1, and Appendix B	Section 2.1.1 of the TSPA-SR Technical Report describes the Implementation of the Features, Events, and Processes Approach including screening of FEPs in sufficient detail to demonstrate the comprehensiveness of the database. The comprehensiveness of the YMP FEP list derives from (a) the diverse backgrounds of the international waste disposal programs contributing to the list, (b) the variety of methods used to identify FEPs including expert judgment, informal elicitation, event tree analysis, stakeholder review, and regulatory stipulation, (c) the continuous iterative discussions and reviews (i.e., at technical workshops and in AMRs) of important YMP attributes, factors, and model components, (d) the systematic and comprehensive classification structure (as described in Step 2 below) ensures that no relevant subject area is overlooked, and (e) the fact that FEPs cannot be removed from the list; they can only be screened out (excluded) from the analysis. Section B.2 of Appendix B provides a detailed description of the process employed to ensure the list of features, events, and processes is comprehensive.
Classification of Processes and Events		
1) DOE has provided adequate documentation identifying how its initial list of processes and events has been grouped into categories.	TSPA-SR Technical Report Section 2.1.1 and Appendix B, FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describes the process and criteria for assigning FEPs as either a primary, secondary, or classification entry. The classification approach adopted for TSPA-SR produced an aggregated set of primary FEPs for screening that covered all identified potentially relevant FEPs for screening potentially relevant Yucca Mountain FEPs. The FEP AMRs provide documentation and justification for screening arguments and dispositions. Documentation is maintained of all mapping of FEPs into primary and secondary categories. For comprehensiveness, traceability is maintained from the secondary to the related primary FEPs.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
2) Categorization of processes and events is compatible with the use of categories during the screening of processes and events.	TSPA-SR Technical Report Section 2.1.1 and Appendix B (Section B.3.2), FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the process and criteria for assigning FEPs as either a primary, secondary, or classification entry. The classification approach adopted for TSPA-SR produced an aggregated set of primary FEPs for screening that covered all identified potentially relevant FEPs for screening potentially relevant Yucca Mountain FEPs. Section B.3.2 describes the objective of the categorization into primary, secondary and classification entries to identify a subset of FEP entries, the primary FEPs, which capture all the issues relevant to the postclosure performance of the potential Yucca Mountain repository and that can be addressed at an appropriate level of screening. The FEP AMRs provide documentation and justification for screening arguments and dispositions. Documentation is maintained of all mapping of FEPs into primary and secondary categories. For comprehensiveness, traceability is maintained from the secondary to the related primary FEPs.
Screening of Processes and Events		
1) Categories of processes and events that are not credible for the YM repository because of waste characteristics, repository design, or site characteristics are identified, and sufficient justification is provided for DOE's conclusions.	TSPA-SR Technical Report 2.1.1 and Appendix B, FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the screening of FEPs for inclusion or exclusion in the TSPA. The FEP screening was performed by subject matter experts and documented in FEP AMRs and are listed in Appendix B of the TSPA-SR Technical Report. Specific guidelines for the basis for screening decisions and the content of screening documentation are also outlined in Appendix B. These guidelines were established to ensure that the screening basis and content for each primary FEP was sufficient to satisfy the screening criteria for low probability, low consequence, or regulatory exclusion. The FEP AMRs provide documentation and justification for screening arguments and TSPA dispositions. Documentation includes a statement of the screening decision for each FEP. Justification is provided for each excluded FEP including the criterion on which it was excluded and the technical basis for the screening argument.
2) The probability assigned to each category of processes and events not screened based on criterion T1 or criterion T2 is consistent with site information, well documented, and appropriately considers uncertainty.	TSPA-SR Technical Report 2.1.1 and Appendix B, FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the screening of FEPs for inclusion or exclusion in the TSPA based on three criteria; regulatory, probability, and consequence. The probability is quantified where possible, although non-quantitative, low-probability arguments are acceptable for "not credible" FEPs. The FEP AMRs provide documentation and justification for screening arguments and dispositions. Probability estimates for FEPs are based on technical analysis of the past frequency of similar events consistent with site information, well documented, and appropriately consider uncertainty.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
3) DOE has demonstrated that processes and events screened from the PA on the basis of their probability of occurrence, have a probability of less than one chance in 10,000 of occurring over 10,000 years.	TSPA-SR Technical Report 2.1.1 and Appendix B (Section B.4.2), FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the screening of FEPs for inclusion or exclusion in the TSPA based on three criteria; regulatory, probability, and consequence. Appendix B, Section B.4.2 describes the FEP database, including the documentation required for screening decisions and screening arguments. FEPs that have less than one chance in 10,000 of occurring over 10,000 years may be excluded (screened out) from the TSPA on the basis of low probability. The FEP AMRs provide documentation and justification for screening arguments and TSPA dispositions. Justification is provided for each excluded FEP including the criterion on which it was excluded and the technical basis for the screening argument. For excluded FEPs, documentation includes the criterion on which it was excluded and the technical basis for the screening argument.
4) DOE has demonstrated that categories of processes and events omitted from the PA on the basis that their omission would not significantly change the calculated expected annual dose, do not significantly change the calculated expected annual dose.	TSPA-SR Technical Report 2.1.1 and Appendix B (Section B.4.2), FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the screening of FEPs for inclusion or exclusion in the TSPA based on three criteria; regulatory, probability, and consequence. Appendix B, Section B.4.2 describes the FEP database including the documentation required for screening decisions and screening arguments. Low consequence exclusions include an explicit statement that there is "no significant change in the expected annual dose." The change in expected annual dose is qualified where possible, and the interpretation of significant change must be described. It is acceptable to quantify the change in an intermediate performance measure, however the qualitative link to change in expected annual dose must be explicitly stated. The FEP AMRs supporting this section provide documentation and justification for screening arguments and TSPA dispositions. For omitted categories, documentation includes the criterion on which it was excluded and the technical basis for the screening argument.
Formation of Scenarios		
1) DOE has provided adequate documentation identifying: (i) whether processes and events have been addressed through consequence model abstraction or scenario analysis and (ii) how the remaining categories of processes and events have been combined into scenario classes.	TSPA-SR Technical Report 2.1.1 and Appendix B, FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the formation of scenario classes and scenario development. All primary FEPs not screened out were retained for inclusion in one or both of the scenario classes. The two scenario classes identified for TSPA-SR, nominal and igneous disruption, are broadly defined and mutually exclusive. The FEP AMRs provide documentation and justification for screening arguments and TSPA dispositions. FEPs that have not been excluded are identified as either expected FEPs or disruptive FEPs. Expected FEPs are included in the TSPA-SR nominal scenario, which is simulated by the base case model described in the TSPA-SR Technical Report documentation. Disruptive scenarios are constructed from expected FEPs and combinations of disruptive FEPs.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
2) The set of scenario classes is mutually exclusive and complete.	TSPA-SR Technical Report 2.1.1 and Appendix B, FEP AMRs	Section 2.1.1 and Appendix B of the TSPA-SR Technical Report describe the formation of scenario classes and scenario development. All primary FEPs not screened out were retained for inclusion in one or both of the scenario classes. The two scenario classes identified for TSPA-SR, nominal and igneous disruption, are broadly defined and mutually exclusive. The FEP AMRs provide documentation and justification for screening arguments and TSPA dispositions. In addition, the AMRs describe the development of the FEPs database, including a description of the construction and screening of scenarios.
Screening of Scenario Classes		
1) Scenario classes that are not credible for the YM potential repository because of waste characteristics, repository design, or site characteristics, individually or in combination, are identified and sufficient justification is provided for DOE's conclusions.	TSPA-SR Technical Report Section 2.1.1	TSPA-SR Technical Report Section 2.1.1 provides justification for screening arguments and TSPA disposition. Scenarios are screened using the same regulatory, probability, and consequence criteria used for screening individual FEPs. For TSPA-SR, scenario screening criteria were evaluated, but all scenario classes were retained.
2) The probability assigned to each scenario class is consistent with site information, well documented, and appropriately considers uncertainty.	TSPA-SR Technical Report Section 2.1.1	The TSPA-SR Technical Report Section 2.1.1 provides justification for screening arguments and TSPA disposition. Probability estimates for scenario classes are based on analyses similar to probabilities assigned for individual FEPs. For TSPA-SR, scenario screening criteria were evaluated, but all scenario classes were retained.
3) Scenario classes that combine categories of processes and events may be screened from the PA on the basis of their probability of occurrence, provided: (i) the probability used for screening the scenario class is defined from combinations of initiating processes and events and (ii) DOE has demonstrated that they have a probability of less than one chance in 10,000 of occurring over 10,000 years.	TSPA-SR Technical Report Section 2.1.1	The TSPA-SR Technical Report Section 2.1.1 describes screening of scenario classes. For TSPA-SR, scenario screening criteria were evaluated, but all scenario classes were retained.
4) Scenario classes may be omitted from the PA on the basis that their omission would not significantly change the calculated expected annual dose, provided DOE has demonstrated that excluded categories of processes and events would not significantly change the calculated expected annual dose.	TSPA-SR Technical Report Section 2.1.1	The TSPA-SR Technical Report Section 2.1.1 describes screening of scenario classes. For TSPA-SR, scenario screening criteria were evaluated, but all scenario classes were retained.
SUBISSUE 3 - Model Abstraction		
Engineered Barrier Degradation	Related to KTI Subissues CLST1, CLST2, CLST6, ENFE2, RDTME3, TEF1, TEF2[1]	

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T1) Sufficient data (field, laboratory or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the WP corrosion abstraction in the TSPA-SR. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA-SR.	TSPA-SR Technical Report Section 3.4, WPD PMR, DE PMR	Section 3.4 of the TSPA-SR Technical Report describes waste package and drip shield degradation. Waste package and drip shield degradation are described in detail in the Waste Package Degradation (WPD) PMR. The WPD PMR describes how the TSPA model requires estimates of corrosion and threshold potentials, both of which are determined through electrochemical testing. Two-year test data has become available and is included in the WPD PMR. Section 3.4.2 of the TSPA-SR Technical Report describes implementation of the model. The computer implementation of the conceptual model provides a mechanism for incorporating the effects of the individual corrosion models in a probabilistic framework that captures the variability and uncertainty in the model parameters. Section 3.4.1 describes output data including the uncertainty and spatial variation of the degradation information on both a waste package and drip shield basis and at different locations within the repository. In the Disruptive Event (DE) AMRs, analog data are described and used as appropriate. The expert elicitations summarized in the framework AMRs were the source of the majority of data used, and data from other sources was qualified as described in the individual AMRs. The process followed for the expert elicitations ensured that relevant data were provided to the experts for consideration.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the WP corrosion abstraction, such as critical relative humidity (RH), material properties, pH, and chloride concentration are technically defensible and reasonably account for uncertainties and variabilities.	TSPA-SR Technical Report Section 3.4, WPD PMR, DE PMR	Section 3.4 of the TSPA-SR Technical Report describes waste package and drip shield degradation. Section 3.4.1 provides a description and illustration of model data sources, input data used by the model and output data generated by the model, including the uncertainty and spatial variation of the degradation information on both a waste package and drip shield. Waste package and drip shield degradation are described in detail in the WPD PMR. The WPD PMR and supporting AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews by a single review team to help ensure consistency. These measures provide confidence that consistency is maintained among the various TSPA models. The process followed for the expert elicitations, as described in disruptive event AMRs ensured that conditions described in this acceptance criterion were met. The methodology to ensure meeting this criterion for data from other sources used in DE AMRs is described in each AMR in Sections 4, 5, and 6 where parameter values, ranges, distributions and bounding assumptions are described. AMRs list assumptions and justify data values, ranges and distributions.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the WP corrosion abstraction.	TSPA-SR Technical Report Section 3.4, WPD PMR, DE PMR	Alternatives models have been considered for rates of dry oxidation, localized corrosion thresholds, stress corrosion thresholds, stress corrosion cracking, stress mitigation, and hydrogen induced cracking. These alternatives are summarized in Section 3.1.11 of the WPD PMR. The DE AMRs discuss alternative conceptual models and data values and ranges consistent with current scientific understanding and justify use of the conceptual models selected. In addition, significant alternative conceptual models are discussed in Section 4 of the DE PMR.
T4) WP corrosion abstraction output is justified through comparison to output of detailed process models or empirical observations (laboratory testings, natural analogs, or both).	TSPA-SR Technical Report Section 3.4, WPD PMR	Model validation, the process through which independent measurements are used to ensure that a model accurately predicts an alteration behavior of waste package materials under a given set of environmental conditions (e.g. under repository environment over the time periods required), is discussed in detail in Section 3.1.10 of the WPD PMR. Output is compared to experimental measurements used as the basis of calculations to verify that correct and reasonable results are obtained.
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the WP corrosion abstraction.	TSPA-SR Technical Report Section 3.4, WPD PMR, DE PMR	The WPD AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models including the WPD corrosion abstraction.
Mechanical Disruption of Engineered Barriers	Related to KTI Subissues RDTME2, RDTME3, CLST1, CLST2, CLST6, IA2, SDS1, SDS2, SDS3, SDS4[1]	
T1) Sufficient data (field, laboratory or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing mechanical disruption of the engineered barriers abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 4.2, DE PMR	TSPA-SR Technical Report Section 4.2 describes the results of the TSPA-SR analysis for igneous disruption. Supporting documentation, including the Disruptive Event (DE) PMR and supporting AMRs, provide details regarding DE models and abstractions. In all DE AMRs, analog data are described and used as appropriate. Expert elicitations were the source of the majority of data used, and data from other sources was qualified as described in the individual DE AMRs. The process followed for the expert elicitations ensured that relevant data were provided to the experts for consideration.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the mechanical disruption of the engineered barriers abstraction, such as probabilistic seismic hazard curves, probability of dike intrusion, and the probability and amount of fault displacement, are technically defensible and reasonably account for uncertainties and variabilities.	DE PMR	The process followed for the expert elicitations ensured that these acceptance criteria were met. The methodology to ensure meeting this criterion for data from other sources used in DE AMRs is described in each AMR in Sections 4, 5 and 6 where parameter values, ranges, distributions and bounding assumptions are described. AMR originators were required to list assumptions and to justify data values, ranges and distributions.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the mechanical disruption of the engineered barriers abstraction.	DE PMR	Section 6 of the AMRs discusses alternative conceptual models and data values and ranges consistent with current scientific understanding and justifies use of the conceptual models selected. In addition, significant alternative conceptual models as presented in NRC IRSRs are discussed in Section 4 of the DE PMR.
T4) Mechanical disruption of the engineered barriers abstraction output is justified through comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	DE PMR	Validation of model outputs is an activity performed by TSPA-SR analysis. All DE AMRs provide documentation of analyses that may be used when comparison with process models and empirical observations is required.
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the mechanical disruption of the engineered barriers abstraction.	DE PMR	The AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models.
Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms	Related to KTI Subissues CLST1, CLST3, CLST4, CLST6, ENFE1, ENFE2, ENFE3, RDTME3, TEF1, TEF2, USFIC4[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the quantity and chemistry of water contacting waste packages and waste forms abstraction in a TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.2, 3.4, 3.5, WFD PMR, UZFT PMR	The basic conceptual model for seepage is described in Section 3.2.4 of the TSPA-SR Technical Report. Section 3.2 also addresses the treatment of uncertainty and variability for each of the components of the UZ model. Sections 3.4 and 3.5 of the TSPA-SR Technical Report include descriptions of the waste package and waste form models. In addition, Section 3.9 of the UZFT PMR summarizes the available data and conceptual models for seepage into drifts. The UZFT PMR describes the effects of coupled processes and their impact on chemical conditions. Section 2.2 describes data collection and Section 3.6 describes the development of properties for the UZ Flow and Transport Model and summarizes available data used to define relevant parameters and conceptual models. The Waste Form Degradation (WFD) PMR summarizes the technical bases of models and corresponding parameters that are abstracted for TSPA-SR. The WFD PMR has been developed in according with applicable QA procedures (see WFD PMR Section 1). Sufficient data were collected by the project or were available in the literature to develop defensible bounding models of the chemical environment. The specific aspects of the water chemistry are discussed in Section 3.2 of the WFD PMR and in supporting AMRs, *In-Package Chemistry Abstraction* (CRWMS M&O 2000 [129287] and *Summary of In-Package Chemistry for Waste*. The UZFT and WFD PMRs have been developed in accordance with applicable QA procedures (see UZFT PMR Section 1.3) for documenting data, analysis, models, and computer code; and preparing and reviewing technical reports.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the quantity and chemistry of water contacting waste packages and waste forms abstraction, such as the pH, carbonate concentration, chloride concentration, and amount of water flowing in and out of the breached WP, are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.2, 3.4, 3.5, WFD PMR, EBS PMR, UZFT PMR	Sections 3.4 and 3.5 of the TSPA-SR Technical Report include descriptions of the model inputs and outputs as well as implementation of the models into the TSPA model. Variations in water chemistry and their effects on in-package chemistry are discussed in Section 3.2 of the WFD PMR. In addition, as described in Section 3.4 of the WFD PMR, a specific model component was added to the total system performance assessment for site recommendation to reasonably account for uncertainties and variabilities in chemistry of water contacting the waste forms. The quantity of water entering the waste package was evaluated in the EBS PMR. The UZFT PMR (Section 3.6) describes the development of properties for the UZ Flow and Transport Model and summarizes available data used to define relevant parameters and conceptual models. Section 3.2 addresses the treatment of uncertainty and variability for each of the components of the UZ model. Also, in the UZFT PMR, Sections 3.7.2 through 3.7.4 summarize UZ flow models and Section 3.7.5 summarizes the results of abstractions of UZ flow.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the quantity and chemistry of water contacting WPs and waste forms abstraction.	TSPA-SR Technical Report Section 3.2, 3.4, 3.5, WFD PMR, UZFT PMR	Each of the sections describing the components of the WFD Model of the WFD PMR describes alternative modeling approaches that could have been used. In the UZFT PMR, alternative conceptual models are discussed in various sections of Chapter 3, "UZ Flow and Transport Model and Abstractions."
T4) Output of quantity and chemistry of water contacting WPs and waste forms abstraction are supported by comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.2, 3.4, 3.5, WFD PMR, UZFT PMR, *In-Package Chemistry Component* AMR	The WFD PMR summarizes the current technical basis of models. For most modeling components within the WFD PMR, a detailed process component model of the phenomena was not developed. Rather, a simplified (abstraction) component was directly developed from the experimental observations and information. For the AMR *In-Package Chemistry Component*, a detailed process model was developed, and then the numerical results used directly through regression to develop a simple empirical relationship. In the UZFT PMR, various sections describe different abstractions and the use of corroborative evidence. Section 3.7.5 describes UZ Flow abstractions for TSPA-SR and how the abstraction of flow was based on detailed process models.
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the quantity and chemistry of water contacting WPs and waste forms abstraction.	TSPA-SR Technical Report Section 3.2, 3.4, 3.5, WFD PMR, UZFT PMR	The WFD AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
Radionuclide Release Rates and Solubility Limits	Related to KTI Subissues CLST3, CLST4, CLST5, CLST6, ENFE3, ENFE4, ENFE5[1]	
T1) Sufficient data (field, laboratory or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing RN release rates and solubility limits abstracted in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.5, WFD PMR	Section 3.5 of the TSPA-SR Technical Report describes the Waste Form Degradation model. This model quantitatively describes the interrelationships among the in-package water chemistry, the degradation of the waste form (including cladding), and the mobilization of radionuclides. This section discusses the treatment of uncertainty and variability in the waste form model. Per the WFD PMR, sufficient data were collected by the project or available in the literature to develop defensible bounding models of the chemical environment. Specific aspects of the radioisotope release rates are discussed in Sections 3.3, 3.4, 3.5, and 3.6 of the WFD PMR and corresponding AMRs. Specific aspects of the solubility limits are discussed in Section 3.7.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the RN release rates and solubility limits abstraction, such as the pH, temperature, colloidal release, and amount of liquid contacting the waste forms, are technically defensible and reasonably account for uncertainties and variabilities.	TSPA-SR Technical Report Section 3.5, WFD PMR, *In-Package Chemistry Component* AMR	Section 3.5 of the TSPA-SR Technical Report describes the Waste Form Degradation model including discussions on the treatment of uncertainty and variability in the waste form model. The WFD PMR summarizes the technical bases of model parameters. Specific aspects of the radioisotope release rates are discussed in Sections 3.3, 3.4, 3.5, 3.6 and corresponding AMRs. Specific aspects of the solubility limits are discussed in Section 3.7 and the corresponding AMRs. To better characterize the uncertainty in the radionuclide release rates and solubility limits, the corresponding modeling components were directly coupled with the AMR *In-Package Chemistry Component* (Section 3.2). In turn, the AMR *In-Package Chemistry Component* uses the available data to couple temporal thermal effects (waste temperature), temporal hydrologic effects (seepage into the package), and temporal chemical effects (degradation rates of steel, aluminum, HLW, SNF) to evaluate the chemical environment inside the WPD.
T3) Alternative waste form dissolution and RN release modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the RN release rates and solubility limits abstraction.	TSPA-SR Technical Report Section 3.5, WFD PMR	In the WFD PMR, each of the sections describing the components of the WFD Model describes alternative modeling approaches that were considered.
T4) RN release rates and solubility limits abstraction output is supported by comparison to outputs of detailed process models or empirical observations (field, laboratory, or natural analog data).	TSPA-SR Technical Report Section 3.5, WFD PMR	The WFD PMR summarizes the current technical basis of models. For most modeling components within the WFD PMR, a detailed process component model of the phenomena was not developed. Rather, a simplified (abstraction) component was directly developed from the experimental observations and information. For radionuclides of neptunium, americium, and uranium, a detailed process model was developed and then the numerical results used directly through regression to develop a simple empirical relationship, as summarized in Sections 3.2 and 3.7. Although material analog data were not used directly, the paragenetic sequence observed in experiments mirrors natural analogs.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the RN release rates and solubility limits abstraction.	TSPA-SR Technical Report Section 3.5, WFD PMR	The WFD AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models.
Spatial and Temporal Distribution of Flow	Related to KTI Subissues ENFE1, RDTME3, SDS2, SDS3, TEF1, TEF2, USFIC1, USFIC2, USFIC3, USFIC4[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the spatial and temporal distribution of flow abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.2, UZFT PMR	Section 3.2 of the TSPA-SR Technical Report describes the Unsaturated Zone Flow Model, including implementation into the TSPA model. Section 3.2 also addresses the treatment of uncertainty and variability for each of the components of the UZ model. The UZFT PMR (Section 3.6) describes the development of properties for the UZ Flow and Transport Model and summarizes available data used to define relevant parameters and conceptual models. The UZFT PMR has been developed in accordance with applicable QA procedures (see UZFT PMR Section 1.3) for documenting data, analysis, models, and computer code, and preparing and reviewing technical reports.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the spatial and temporal distribution of flow abstraction (such as the effects of climate change on infiltration, near surface influences [e.g., evapotranspiration and runoff] on infiltration, structural controls on the spatial distribution of deep percolation, and thermal reflux owing to repository heat load) are technically defensible and reasonably account for uncertainties and variabilities.	TSPA-SR Technical Report Section 3.2, UZFT PMR	The UZFT PMR (Section 3.6) describes the development of properties for the UZ Flow and Transport Model and summarizes available data used to define relevant parameters and conceptual models. Section 3.2 addresses the treatment of uncertainty and variability for each of the components of the UZ model. Also, in the UZFT PMR, Sections 3.7.2 through 3.7.4 summarize UZ flow models, and Section 3.7.5 summarizes the results of abstractions of UZ flow.
T3) Alternative modeling approaches, consistent with available data and current scientific understanding, are investigated and results and limitations appropriately factored into the spatial and temporal distribution of flow abstraction.	TSPA-SR Technical Report Section 3.2, UZFT PMR	In the UZFT PMR, alternative conceptual models are discussed in various sections of Chapter 3, "UZ Flow and Transport Model and Abstractions."

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T4) Spatial and temporal distribution of flow abstraction output is justified through comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.2, UZFT PMR	Section 3.2 of the TSPA-SR Technical Report describes the Unsaturated Zone Flow model including implementation into the TSPA model. Section 3.5.3 describes abstractions of climate and infiltration, and Section 3.2.3.2 describes implementation of the Mountain Scale Flow model into the TSPA. Climate and infiltration are not included directly in TSPA simulations, but only indirectly through its use as a boundary condition for the UZ flow and thermal hydrology models. In the UZFT PMR, various sections describe different abstractions and the use of corroborative evidence. Section 3.7.5 describes UZ Flow abstractions for TSPA-SR and how the abstraction of flow was based on detailed process models.
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the spatial and temporal distribution of flow abstraction.	TSPA-SR Technical Report Section 3.2, UZFT PMR	The UZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. In the UZFT PMR, the abstraction of UZ flow is summarized in Section 3.7.5.
Flow Paths in the Unsaturated Zone	Related to KTI Subissues ENFE1, SDS3, TEF1, TEF2, USFIC4[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the flow paths in the UZ in the abstraction in TSPA. Where adequate data cannot be readily obtained, other information sources such as expert elicitation or bounding values have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.2, UZFT PMR	Section 3.2 of the TSPA-SR Technical Report describes the Unsaturated Zone Flow Model, including implementation into the TSPA model. Section 3.2 also addresses the treatment of uncertainty and variability for each of the components of the UZ model. The UZFT PMR (Section 3.6) describes the development of properties for the UZ Flow and Transport Model and summarizes available data used to define relevant parameters and conceptual models. The UZFT PMR has been developed in accordance with applicable QA procedures (see UZFT PMR Section 1.3) for documenting data, analysis, models, and computer code, and preparing and reviewing technical reports.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the flow paths in the UZ in the abstraction, such as hydrologic properties, stratigraphy, and infiltration rate, are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.2, UZFT PMR	The UZFT PMR (Section 3.6) describes the development of properties for the UZ Flow and Transport Model and summarizes available data used to define relevant parameters and conceptual models. Section 3.2 addresses the treatment of uncertainty and variability for each of the components of the UZ model. Also, in the UZFT PMR, Sections 3.7.2 through 3.7.4 summarize UZ flow models, and Section 3.7.5 summarizes the results of abstractions of UZ flow.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the distribution on mass flux between fracture and matrix in the abstraction.	TSPA-SR Technical Report Section 3.2, UZFT PMR	In the UZFT PMR, alternative conceptual models are discussed in various sections of Chapter 3, "UZ Flow and Transport Model and Abstractions."

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T4) Flow paths in the UZ abstraction output are justified through comparison to output of detailed flow process models or empirical observations (laboratory testings, natural analogs, or both).	TSPA-SR Technical Report Section 3.2, UZFT PMR	Section 3.2 of the TSPA-SR Technical Report describes the Unsaturated Zone Flow model, including implementation into the TSPA model. Section 3.5.3 describes abstractions of climate and infiltration. Section 3.2.3.2 describes implementation of the Mountain Scale Flow model into the TSPA. This section describes that direct use of the 3-D mountain-scale flow model eliminates the need to test simplified abstractions against more complex models. In the UZFT PMR, various sections describe different abstractions and the use of corroborative evidence. Section 3.7.5 describes UZ Flow abstractions for TSPA-SR and how the abstraction of flow was based on detailed process models. In the UZFT PMR, the abstraction was based on detailed process models described in Sections 3.7.2 through 3.7.4.
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the flow paths in the UZ abstraction.	TSPA-SR Technical Report Section 3.2, UZFT PMR	The UZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. In the UZFT PMR, the abstraction of UZ flow is summarized in Section 3.7.5.
Radionuclide Transport in the Unsaturated Zone	Related to KTI Subissues RT1, RT3, RT4, SDS3, ENFE4, USFIC4, USFIC6[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the spatial and temporal distribution of flow abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.7, UZFT PMR	Section 3.7 of the TSPA-SR Technical Report describes the Unsaturated Zone Transport Model including implementation into the TSPA model and the treatment of uncertainty and variability. In the UZFT PMR, Section 3.11.3 summarizes the transport properties that are used for the abstraction of radionuclide transport in the UZ. The AMR on UZ and SZ transport properties provides additional detail on transport properties for the UZ. The UZFT PMR has been developed in accordance with applicable QA procedures (see UZFT PMR Section 1.3) for documenting data, analysis, models, and computer code, and preparing and reviewing technical reports.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the spatial and temporal distribution of flow abstraction (such as the effects of climate change on infiltration, near surface influences [e.g., evapotranspiration and runoff] on infiltration, structural controls on the spatial distribution of deep percolation, and thermal reflux owing to repository heat load) are technically defensible and reasonably account for uncertainties and variabilities.	TSPA-SR Technical Report Section 3.7, UZFT PMR	In the UZFT PMR, Section 3.11.3 summarizes parameter values, ranges, distributions, and bounding assumptions for the abstraction of radionuclide transport, as applicable. The UZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, Section 3.7.2 of the TSPA-SR Technical Report describes the implementation of the UZ Transport Model into the TSPA. This section describes the connections between UZ transport and other TSPA model components. The UZ transport model is directly coupled with the TSPA model.
T3) Alternative modeling approaches, consistent with available data and current scientific understanding, are investigated and results and limitations appropriately factored into the RT in the UZ abstraction.	TSPA-SR Technical Report Section 3.7, UZFT PMR	In the UZFT PMR, Section 3.11.9 summarizes an alternative conceptual model for transport in the UZ. An alternative conceptual model that does not allow diffusion (but still allows advection) into the matrix was investigated. The results are described in the section.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T4) RT in the UZ abstraction output is justified through comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.7, UZFT PMR	In the UZFT PMR, Section 3.11.13 describes the abstraction of radionuclide transport. Section 3.11.13.4 describes how the abstraction is compared to alternate solution methods for a variety of problems.
T5) Important design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the consideration of RT in the UZ abstraction.	TSPA-SR Technical Report Section 3.7, UZFT PMR	The UZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. In the UZFT PMR, Section 3.11.13 discusses the abstraction of radionuclide transport, including physical phenomena, couplings, and assumptions.
Flow Paths in the Saturated Zone	Related to KTI Subissues USFIC2, USFIC5, USFIC6, SDS1, SDS3[1]	
T1) Sufficient hydrogeologic data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the flow paths in the SZ abstraction in the TSPA-SR. Where adequate data does not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.8, SZFT PMR	Section 3.8 of the TSPA-SR Technical Report describes the Saturated Zone Flow Model, including implementation into the TSPA model. In addition, Section 3.8.1.3 addresses the treatment of uncertainty and variability for the SZ flow model. In the SZFT PMR, characterization and site data are discussed in Section 3.1; conceptual models are discussed in Section 3.2; and synthesis of SZ model and model abstractions are discussed in Section 3.6. Also, the SZFT PMR has been developed in accordance with applicable QA procedures (see SZFT PMR Section 1.4) for documenting data, analysis, models, and computer code, and preparing and reviewing technical reports.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the flow paths in the SZ abstraction, such as the effect of climate change on the SZ fluxes and water table level and well pumping practices, are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.8, SZFT PMR	Section 3.8 of the TSPA-SR Technical Report describes the Saturated Zone Flow Model. Section 3.8.1 describes the relationship between the SZ flow and other components of the TSPA. Section 3.8.1.3 addresses the treatment of uncertainty and variability for the SZ flow model. The SZFT PMR and supporting AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews by a single review team to help ensure consistency. These measures provide confidence that consistency is maintained among the various TSPA models.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the flow paths in the SZ.	TSPA-SR Technical Report Section 3.8, SZFT PMR	In the SZFT PMR, alternative conceptual models are discussed in Section 3.2.5.
T4) Flow paths in the SZ abstraction output are justified through comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.8, SZFT PMR	In the SZFT PMR, the process of testing the validity of the conceptual, mathematical and numeric representation of the system is discussed in Section 3.4. Also, analog studies that have been conducted in saturated environments as a means of model "validation" or confidence building are discussed in Section 3.4.5 of the PMR.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T5) Important site (geologic and hydraulic) features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the flow paths in the SZ abstraction.	TSPA-SR Technical Report Section 3.8, SZFT PMR	The SZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. In the SZFT PMR, synthesis of SZ model and model abstractions is discussed in Section 3.6 and assumptions, uses, and limitations of models are discussed in Section 3.5.
Radionuclide Transport in the Saturated Zone	Related to KTI Subissues RT1, RT2, RT3, RT4, SDS3, USFIC5, USFIC6[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the RT in the SZ abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.8, SZFT PMR	Section 3.8 of the TSPA-SR Technical Report describes the Saturated Zone Flow Model, including implementation into the TSPA model and a description of the data that supports radionuclide transport in the SZ. Section 3.8.2.4 addresses the treatment of uncertainty and variability for each of the components of the SZ model. In the SZFT PMR, SZ characterization and data is discussed in Section 3.1. Section 3.1.3 provides a summary of hydrochemical data pertinent to transport. Also, dilution in water supply in abstraction of radionuclide transport is discussed in Section 3.6.3, and synthesis of SZ model and model abstractions is discussed in Section 3.6. The SZFT PMR has been developed in accordance with applicable QA procedures (see SZFT PMR Section 1.4) for documenting data, analysis, models, and computer code, and preparing and reviewing technical reports.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the RT in the SZ abstraction, such as the sorption on fracture surfaces and Kd for matrix, are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.8, SZFT PMR	Section 3.8 of the TSPA-SR Technical Report describes the Saturated Zone Flow Model including SZ transport. Section 3.8.2 describes the data that supports radionuclide transport and describes the relationship between the SZ flow and other components of the TSPA. Section 3.8.2.4 addresses the treatment of uncertainty and variability for the SZ flow model. The SZFT PMR and supporting AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews by a single review team to help ensure consistency. These measures provide confidence that consistency is maintained among the various TSPA models. In the SZFT PMR, SZ characterization and data are discussed in Section 3.1; dilution in water supply abstraction of radionuclide transport is discussed in Section 3.6.3; parameter uncertainty distributions are discussed in Section 3.7.2; probabilistic analyses are discussed in Section 3.7.3; and syntheses of SZ model and model abstractions are discussed in Section 3.6.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the RT in the SZ abstraction.	TSPA-SR Technical Report Section 3.8, SZFT PMR	In the SZFT PMR, alternative conceptual models are discussed in Section 3.2.5.
T4) RT in the SZ abstraction output is justified through comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.8, SZFT PMR	In the SZFT PMR, the process of testing the validity of the conceptual, mathematical and numeric representation of the system is discussed in Section 3.4. Also, analog studies that have been conducted in saturated environments as a means of model "validation" or confidence building are discussed in Section 3.4.5 of the PMR.
T5) Important physical phenomena and couplings and consistent and appropriate assumptions are incorporated into the consideration of RT in the SZ abstraction.	TSPA-SR Technical Report Section 3.8, SZFT PMR	The SZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. In the UZFT PMR, dilution in water supply in abstraction of radionuclide transport is discussed in Section 3.6.3; assumptions, uses, and limitations of model are discussed in Section 3.5; and synthesis of SZ model and model abstractions is discussed in Section 3.6.
Volcanic Disruption of Waste Packages	Related to KTI Subissues IA1, IA2, SDS1, SDS4[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for abstracting the volcanic disruption of WPs in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.10, DE PMR	Section 3.10.1 of the TSPA-SR Technical Report describes the development of the conceptual model for igneous activity at Yucca Mountain. The conceptual model has three main components. Each of the components is based on observations of the past geologic record and, for the characteristics of an eruption, observations of modern analogs. Basing the conceptual model for possible future igneous activity on the past record and modern analogs is consistent with the proposed regulatory requirement to assume that the "(evolution of the geologic setting shall be consistent with present knowledge of natural processes" [proposed 10 CFR 63.115(a)(4)] [64 FR 8640] [101680]). In all DE AMRs, analog data are described and used as appropriate. Expert elicitations were the source of the majority of data used, and data from other sources was qualified as described in individual AMRs. Section 3.10.1.2 describes how the probability of future igneous activity in the Yucca Mountain region used in the TSPA-SR is based on the *Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada* (PVHA) (CRWMS M&O 1996 [100116]) conducted by the DOE in 1995 and 1996. Ten experts in the field of volcanology evaluated available data on past volcanic activity in the region and provided expert judgement on the probability of future igneous activity. Their judgments (elicitations) were combined to produce an integrated assessment of the volcanic hazard that reflects a range of alternative scientific interpretations. The process followed for the expert elicitations ensured that relevant data were provided to the experts for consideration.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the volcanic disruption of WPs abstraction are technically defensible and reasonably account for uncertainties and variability. The technical basis for the parameter values used in the PA needs to be provided.	TSPA-SR Technical Report Section 3.10, DE PMR	Section 3.10 of the TSPA-SR Technical Report describes volcanism and the development of the conceptual model for igneous activity at Yucca Mountain. Each of the components of the model is based on observations of the past geologic records and for the characteristics of the eruption, observations of modern analogs. Information used in the TSPA-SR model to characterize the intrusive and eruptive processes comes from three sources: examination of the geologic record of past intrusive and eruptive events in the Yucca Mountain region; observations of eruptive processes during analogous modern volcanic events elsewhere in the world; and consideration of the range of physical processes that might occur during the interaction between the repository and an igneous dike. Variability in parameters was accounted for through consideration of the range of physical processes that might occur. In addition, DOE bases its TSPA-SR base case on the full range of probability values as described in the igneous framework AMR. This acceptance criterion is specifically addressed in the Igneous Activity IRSR, Probability Acceptance criterion 4. The expert elicitation process was used to ensure that uncertainty in estimates of the volcanic hazard was completely captured. Treatment of uncertainty in the volcanic hazard formulation is described in the PVHA.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the volcanic disruption of WPs abstraction.	TSPA-SR Technical Report Section 3.10, DE PMR, Igneous Consequences Modeling for TSPA-SR AMR	Consideration given to alternative modeling approaches is described in the AMR Igneous Consequences Modeling for TSPA-SR (DE PMR Section 3.1.5). In addition, DE AMRs discuss alternative conceptual models and data values and ranges consistent with current scientific understanding and justify use of the conceptual models selected. In addition, significant alternative conceptual models as presented in NRC IRSRs are discussed in Chapter 4 of the DE PMR.
T4) Outputs of the volcanic disruption of WPs abstraction are justified through comparison to outputs of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.10, DE PMR	Section 3.10.2 of the TSPA-SR Technical Report describes the TSPA-SR model for volcanic eruption. This section describes how entrainment of waste and atmospheric transport of contaminated ash is modeled using the ASHPLUME code. Use of the ASHPLUME model is considered reasonable as discussed in the igneous consequences AMR. Additional analyses of the suitability of ASHPLUME to model tephra deposits at Cerro Negro based on the 1995 eruption have been completed. The results of the analysis are in good agreement with the observed ash distribution, and the agreement is considered to verify the utility of the model.
T5) Important site and design features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the volcanic disruption of WPs abstraction and the technical bases are provided.	TSPA-SR Technical Report Section 3.10, DE PMR	The DE AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
Airborne Transport of Radionuclides	Related to KTI Subissues ISI, IA2[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the airborne transport of RNs abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.10, DE PMR	Section 3.10 of the TSPA-SR Technical Report describes volcanism and the development of the conceptual model for igneous activity at Yucca Mountain. Direct observations of volcanic processes in the subsurface are extremely rare and there are no precedents for modeling. Information used in the TSPA-SR model comes from three sources: examination of the geologic record of past intrusive and eruptive events in the Yucca Mountain region; observations of eruptive processes during analogous modern volcanic events elsewhere in the world; and consideration of the range of physical processes that might occur during the interaction between the repository and an igneous dike. Also, in all DE AMRs, analog data are described and used as appropriate. Expert elicitations are the source of the majority of data used, and data from other sources was qualified as described in individual AMRs. The process followed for the expert elicitations ensured that relevant data were provided to the experts for consideration.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the airborne transport of RNs abstraction, such as the magnitude of eruption and deposition velocity, are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.10, DE PMR, Biosphere PMR, *Characterize Eruptive Processes at Yucca Mountain, Nevada* AMR, *Igneous Consequences Modeling for TSPA-SR* AMR	The basis for selection of parameter values, such as magnitude of eruption that are inputs to the igneous consequences modeling, are described in the AMR *Characterize Eruptive Processes at Yucca Mountain, Nevada* (DE PMR, Section 3.1.2). Deposition velocities used are described in the AMR *Igneous Consequences Modeling for TSPA-SR* (DE PMR Section 3.1.5). Section 3.10 of the TSPA-SR Technical Report describes how uncertainties and variability are reasonably accounted for through the specification of distributions of reasonably possible values. Information about eruption characteristics, the probability of eruptive conduits forming within the potential repository, and the potential repository response to eruption are used to develop a distribution of parameter values characterizing uncertainty in the extent of damage to waste packages and the amount of waste available to be entrained in the eruption. Modeling yields a distribution of results characterizing uncertainty in the concentration of waste particles on the ground surface. In addition, Section 3.2.4 of the Biosphere PMR discusses input parameters for GENII-S, and Section 3.3.2 discusses the disruptive event biosphere dose conversion factors to support dose consequence calculations in TSPA.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the airborne transport of RNs abstraction.	TSPA-SR Technical Report Section 3.10, DE PMR, *Igneous Consequences Modeling for TSPA-SR* AMR	Consideration given to alternative modeling approaches is described in the AMR *Igneous Consequences Modeling for TSPA-SR* (DE PMR Section 3.1.5). In addition, DE AMRs discuss alternative conceptual models and data values and ranges consistent with current scientific understanding and justify use of the conceptual models selected. In addition, significant alternative conceptual models as presented in NRC IRSRs, are discussed in Chapter 4 of the DE PMR.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T4) Airborne transport of RNs abstraction output is justified through comparison to output of detailed process models or empirical observations (laboratory testing, natural analogs, or both).	TSPA-SR Technical Report Section 3.10, DE PMR	Section 3.10.2.2 of the TSPA-SR Technical Report describes the TSPA-SR model for volcanic eruption. This section describes how entrainment of waste and atmospheric transport of contaminated ash is modeled using the ASHPLUME code. Use of the ASHPLUME model is considered reasonable as discussed in the igneous consequences AMR. Additional analyses of the suitability of ASHPLUME to model tephra deposits at Cerro Negro based on the 1995 eruption have been completed and are being documented. The results of the analysis are in good agreement with the observed ash distribution, and the agreement is considered to verify the utility of the model.
T5) Important site features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the airborne transport of RNs abstraction.	TSPA-SR Technical Report Section 3.10, DE PMR	The DE AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models.
Dilution of Radionuclides Due to Well Pumping	Related to KTI Subissue USFIC5[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the dilution of RNs due to well pumping abstraction in the TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.9, SZFT PMR, Biosphere PMR, *Groundwater Usage by the Proposed Farming Community* AMR	Section 3.9 of the TSPA-SR Technical Report describes the Biosphere conceptual model. Included in this section is a discussion of the treatment of uncertainty and variability. In the SZFT PMR, dilution in water supply in abstraction of radionuclide transport is discussed in Section 3.6.3. The element of dilution of radionuclides in groundwater due to well pumping is addressed in the biosphere process by an alternative approach based on an approximation that removes uncertainty associated with dilution due to pumping dynamics. This approach is consistent with information presented in the statement of consideration to the proposed rule 10 CFR 63. Water usage is discussed in Section 3.4 of the Biosphere PMR. 10 CFR 63.115 provides a reference biosphere and receptor group criterion that allow use of a simple model that avoids speculation associated with detailed dilution modeling. The average annual concentration of the radionuclides in the groundwater is derived by distributing the mass of radionuclides crossing the 20-km boundary annually uniformly over the total annual water usage. The biosphere effort described in this document derived the annual volume of water estimated to be used by the community. The SZFT PMR developed the mass of each radionuclide crossing the 20-km boundary annually. The TSPA code divides the latter by the former to derive the annual average concentration from which the annual expected dose to an average member of the critical group is obtained when multiplied by the BDCFs. In deriving the annual water usage, this element is addressed to the extent possible with the simplistic but conservative dilution model found in the groundwater usage AMR (*Groundwater Usage by the Proposed Farming Community*). In addition, the TSPA assumes that all the mass of radionuclides crossing the 20-km boundary is captured in the pumped water.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the dilution of RNs in groundwater due to well pumping abstraction, such as the pumping well characteristics and water usage by the receptor groups, are technically defensible and account for uncertainty and variability.	TSPA-SR Technical Report Section 3.9, SZFT PMR, Biosphere PMR	Section 3.9 of the TSPA-SR Technical Report describes the Biosphere conceptual model. Included in this section is a discussion of the treatment of uncertainty and variability. The SZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. Effective integration of the Biosphere into the TSPA, as described in Section 3.9.2, also helps ensure the consistent use of values. Also see T1 above for discussion of how the Biosphere PMR addresses acceptance criteria for this subissue.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the dilution of RNs in groundwater due to well pumping abstraction.	TSPA-SR Technical Report Section 3.9, SZFT PMR, Biosphere PMR	See T1 above for discussion of how the Biosphere PMR addresses acceptance criteria for this subissue. In the SZFT PMR, Section 3.8 discusses other views and alternative models.
T4) Dilution of RNs due to well pumping abstraction output is justified through comparison to outputs of detailed process models or empirical observations (laboratory test).	TSPA-SR Technical Report Section 3.9, SZFT PMR, Biosphere PMR	See T1 above for discussion of how the Biosphere PMR addresses acceptance criteria for this subissue. In the SZFT PMR, validation is discussed in Section 3.4.
T5) PA analyses incorporate important hydrogeologic features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the dilution of RNs due to well pumping abstraction.	TSPA-SR Technical Report Section 3.9, SZFT PMR, Biosphere PMR	The Biosphere and SZFT AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. Also see T1 above for discussion of how the Biosphere PMR addresses acceptance criteria for this subissue.
Redistribution of Radionuclides in Soil	Related to KTI Subissue IA2[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models necessary for developing the redistribution of RNs in soil abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	TSPA-SR Technical Report Section 3.10, Biosphere PMR	Section 3.10.3 of the TSPA-SR Technical Report describes processes that must be considered after the ash is deposited including ash resuspension, redistribution and erosion, as well as radionuclide uptake in plants and animals. The Biosphere PMR discusses the abstraction of the soil processes in Section 3.3.1.1. The soil analysis incorporates representative sandy soil Kd values and other parameter values including bulk soil density, precipitation, evaporation, and irrigation rate based on data from the Amargosa Valley to calculate leaching coefficients and the rates of soil removal by erosion. These parameters support the analysis of soil build up and erosion processes affecting the BDCF abstractions for TSPA-SR. The SZFT PMR has been developed in accordance with applicable QA procedures (see SZFT PMR Section 1.4) for documenting data, analysis, models, and computer code, and preparing and reviewing technical reports.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the redistribution of RNs in soil abstraction, such as depth of the plowed layers and mass loading factor, are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.10, Biosphere PMR, *Evaluate Soil/Radionuclide Removal by Erosion and Leaching* AMR, *Abstraction of BDCF Distributions for Irrigation Periods* AMR	Implementing procedures compliant with the QARD controls DOE's development of parameter values, assumed ranges, probability distributions and bounding assumptions. These procedures require measures intended to help ensure technical defensibility and appropriate accounting for uncertainties and variabilities. The AMRs and PMR for the Biosphere PMR have been developed in accordance with these procedures. The *Evaluate Soil/Radionuclide Removal by Erosion and Leaching* AMR addresses soil parameters. The *Abstraction of BDCF Distributions for Irrigation Periods* AMR describes how the build-up due to irrigation and erosion due to other processes are abstracted for use in TSPA code. Section 3.3.1.1 of the Biosphere PMR summarizes the abstraction process.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and their results and limitations appropriately factored into the redistribution of RNs in soil abstraction.	TSPA-SR Technical Report Section 3.10, Biosphere PMR, *Evaluate Soil/Radionuclide Removal by Erosion and Leaching* AMR	The Biosphere AMR, *Evaluate Soil/Radionuclide Removal by Erosion and Leaching,* addresses selection of parameter values and alternative approaches to evaluation of soil erosion and build-up. The analysis of soil processes is coupled with the BDCF abstraction to account for radionuclide build-up/erosion for use in TSPA.
T4) Redistribution of RNs in soil output is justified through comparison to output of detailed process models or empirical observations (laboratory testings, natural analogs, or both).	TSPA-SR Technical Report Section 3.10, Biosphere PMR	Evaluation of soil processes for dilution of radionuclides is based to the extent practical on analog processes. Section 3.2.4.1.2. of the Biosphere PMR discusses soil parameters pertinent to the biosphere model. Abstraction of BDCF distributions for irrigation periods couples the soil erosion processes with irrigation period build-up or radionuclides as discussed in Section 3.3.1.1 of the Biosphere PMR.
T5) Important site features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the redistribution of RNs in soil abstraction.	TSPA-SR Technical Report Section 3.10, Biosphere PMR	The Biosphere AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained among the various TSPA models. Abstraction of BDCF distributions for irrigation periods couples the soil erosion processes with irrigation period build-up or radionuclides as discussed in Section 3.3.1.1 of the Biosphere PMR. Consistent and appropriate assumptions have been applied to analysis of soil process for calculation of BDCFs for use in TSPA.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
Lifestyle of the Critical Group	Related to KTI Subissues USFIC1, USFIC2, USFIC3, USFIC5, RT3, IA2[1]	
T1) Sufficient data (field, laboratory, or natural analog data) are available to adequately define relevant parameters and conceptual models as necessary for developing the lifestyle of critical group abstraction in TSPA. Where adequate data do not exist, other information sources such as expert elicitation have been appropriately incorporated into the TSPA.	Biosphere PMR, TSPA-SR Technical Report Section 3.9	Section 3.9 of the TSPA-SR Technical Report describes the Biosphere. This section includes a description of how FEPs have been screened to identify those that are relevant to the biosphere in the vicinity of Yucca Mountain. Section 3.9.1 provides a description of the definition of the receptor including the characteristics of the receptor, the critical group and the local environment. Section 3.1.2.1 of the Biosphere PMR describes the data used to identify the critical group to support TSPA. The 1997 dietary survey data is used to characterize consumption for the critical group. Bureau of Census data is used to characterize other lifestyle characteristics of the critical group. These data are sufficient to adequately quantify parameters for the critical group. Characteristics of the human receptor related to the key exposure pathways are discussed in Section 3.2.4.2.
T2) Parameter values, assumed ranges, probability distributions, and bounding assumptions used in the lifestyle of critical group abstraction such as consumption rates, plant and animal uptake factors, mass loading factors, and BDCFs are technically defensible and reasonably account for uncertainties and variability.	TSPA-SR Technical Report Section 3.9, Biosphere PMR	Section 3.9 of the TSPA-SR Technical Report describes the Biosphere. This section includes a description of the integration of the Biosphere into the TSPA. Implementing procedures compliant with the QARD controls DOE's development of parameter values, assumed ranges, probability distributions and bounding assumptions. Section 3.2.4 of the Biosphere PMR discusses parameters pertaining to the environmental transport of radionuclides and characteristics of the human receptor. Biosphere AMRs provide a detailed description of parameter value for determination of critical group. Parameter values and associated ranges and distributions considered in the lifestyle of critical group abstractions are defensible and reasonable. The Biosphere AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency.
T3) Alternative modeling approaches consistent with available data and current scientific understanding are investigated and results and limitations appropriately factored into the lifestyle of critical group abstractions.	TSPA-SR Technical Report Section 3.9, Biosphere PMR	The Biosphere PMR conceptual approach to the critical group is qualitative. The DOE guidance for identification of the critical group (Dyer 1999 [105655], enclosure, Section 115[b]) is consistent with the NRC's proposed rule at 10 CFR 63.115(b) (64 FR 8640 [101680]). To select a critical group, individuals likely to be at the highest risk from among the exposed population were specified. An analysis constructed four screening groups that encompass the range of diet, lifestyle, and land use characteristics expected to be found in the region surrounding Yucca Mountain. The screening groups were analyzed against known data. The potential exposure of the screening groups was qualitatively evaluated. The relative ranking, by exposure potential of essential characteristics and attributes of the screening groups was determined. These four screening groups and their associated attributes are discussed in Section 3.1.2.2 of the Biosphere PMR.

Table D.1-4. Total System Performance Assessment and Integration Issue Resolution Status Report Rev. 2 Acceptance Criteria (Continued)

TSPAI IRSR REV. 2 Acceptance Criteria	Related Documentation	Approach
T4) Dose calculation output pertaining to lifestyle of the critical group is justified through comparison to output of detailed process models, and/or empirical observations (field data, laboratory data, or natural analogs).	TSPA-SR Technical Report Section 3.9, Biosphere PMR	Section 3.9.2 of the TSPA-SR Technical Report provides the principal results of the biosphere modeling work. The results are expressed as the amount of radiation dose received annually by the receptor for each unit of radioactivity concentration introduced into the Biosphere. Section 3.3 of the Biosphere PMR describes the development of the biosphere dose conversion factors for input to the TSPA code for calculation of expected annual doses.
T5) Important site features, physical phenomena and couplings, and consistent and appropriate assumptions are incorporated into the lifestyle of the critical group abstraction.	TSPA-SR Technical Report Section 3.9, Biosphere PMR	The Biosphere AMRs document assumptions, important site features, and physical phenomena and couplings. These documents have been subjected to thorough interdisciplinary reviews to help ensure consistency. In addition, the PMRs that summarize and integrate the results of the AMRs have been subjected to a review by a single review team. One of the main objectives of this review was to identify inconsistencies among PMRs. These measures provide confidence that consistency is maintained as appropriate among the various TSPA models.
SUBISSUE 4 - Demonstration of the Overall Performance Objective		
The final requirements for the overall performance objective will be established after the rule is published in final form.		

Table D.1.5. Crosswalk of Key Technical Issues to Process Model Reports

KTI * Subissues	Process Model Report								
	Biosphere[1]	Disruptive Events (DE)[2]	Engineered Barrier System (EBS)[3]	Near-Field Environment (NFE)[4]	SZ Flow and Transport (SZ)[5]	UZ Flow and Transport (UZ)[6]	Waste Form (WF)[7]	Waste Package (WP)[8]	ISM
USFIC1						■			
USFIC2					▨	■			
USFIC3						■			
USFIC4						■			
USFIC5					■				
USFIC6					■	■			
TEF1			▨	■					
TEF2			■	▨		■			
TEF3			■			■			
ENFE1			▨	▨		▨			
ENFE2			■	▨					
ENFE3			■				■		
ENFE4			■	▨		■			
ENFE5	(The subissue of criticality is addressed in a topical report on nuclear criticality and supporting documents.)								
CLST1			▨					■	
CLST2		▨						■	
CLST3			▨				■		
CLST4			▨				■		
CLST5	(The subissue of criticality is addressed in a topical report on nuclear criticality and supporting documents.)								
CLST6		▨	▨					▨	
RT1, Rev 1						▨			
RT2, Rev 1					■				
RT3, Rev 1					▨	■			
RT4, Rev 1	(The subissue of criticality is addressed in a topical report on nuclear criticality and supporting documents.)								
TSPAI1	▨	▨	▨	▨	▨	▨	▨	▨	
TSPAI2	■	■	■	■	■	■	■	■	
TSPAI3	▨	▨	▨	▨	▨	▨	▨	▨	
TSPAI4	(Criteria for this subissue will be established after the rule is published in final form.)								
IA1		■							
IA2	■	■							
SDS1		■							
SDS2		■							
SDS3					▨	■			
SDS4		■							
RDTME1	(Criteria for this subissue are elements of the Design Control part of the Quality Assurance program addressed in Section 3 of the QARD.)								
RDTME2	(This subissue will be addressed in the Seismic Topical Report.)								
RDTME3			▨	▨					
RDTME4	(Criteria for this subissue have not yet been developed.)								

■ The subissue is completely applicable to the PMR.

▨ The subissue is partially applicable to the PMR.

Source: [1]CRWMS M&O 2000 [151615], [2]CRWMS M&O 2000 [141733], [3]CRWMS M&O 2000 [145796], [4]CRWMS M&O 2000 [146589], [5]CRWMS M&O 2000 [145738], [6]CRWMS M&O 2000 [145774], [7]CRWMS M&O 2000 [138332], [8]CRWMS M&O 2000 [138396]

NOTE: QARD = *Quality Assurance Requirements and Description* (DOE 2000 [149540])

D.2 ISSUE RESOLUTION STATUS REPORTS TRACKING DATABASE - A MICROSOFT ACCESS 97 DATABASE

The IRSR Tracking Database was developed by using Microsoft Access 97. This Access database includes tables, queries, forms, reports, switchboard, macro, and modules. The information of IRSR Tracking Database is mainly stored in tables. Query is a tool for search. Forms are for data entry or update. Reports sorted and organized by different elements can also be found in the database.

D.2.1 ACCESS TABLES

There are four key tables included in the database. The Access tables can be exported to Microsoft Excel as well as other databases (e.g., Paradox, dBase).

D.2.1.1 Issue Resolution Status Reports Acceptance Criteria and Integrated Subissues Tables

The IRSR Acceptance Criteria table includes all of the acceptance criteria. The IRSR Acceptance Criteria table is able to link with the IRSR GODO Database that includes all IRSR Acceptance Criteria, NRC review methods, and NRC expectations. Additionally, an ISI table is created to link ISIs with KTI subissues.

D.2.1.2 Repository Safety Strategy 4 Table

The process model factors and FEPs are included in the RSS4 table. Since the RSS4 Workshop provides the list of process model factors and screened FEPs for the nominal scenario, the table is entitled RSS4.

D.2.1.3 Mapping Table

The Mapping table provides a roadmap to link PMR and process model factors with IRSR acceptance criteria. The PMRs responses to applicable IRSR acceptable criteria are also included. By using the "relationship" function provided in Access 97, detailed information of PMR, process model factors, FEPs, and IRSR acceptance criteria from the linked database or tables can be obtained. For example, since both RSS4 and Mapping tables consist of process model factors, the Mapping table can relate to the RSS4 table for FEP data. By using the same relationship function, the Mapping table can relate to IRSR Acceptance Criteria table for the detailed acceptance criteria.

D.2.1.4 Performance Assessment Operations Summary Table

Most of the data in the Performance Assessment Operations summary table are obtained from Total System Performance Assessment-Site Recommendation Methods and Assumptions (CRWMS M&O 1999 [123126]), issued in 1999. The data fields of this table consist of the following:

- Key Attributes of System
- RSS/Process Model Factor

- Process Model Report
- TSPA component
- NRC KTI
- NRC Components of Subsystem
- KESA (revised to ISI)
- Analysis and Model Report
- PA Lead
- IRSR Licensing Lead
- PMR Licensing Lead.

D.2.2 ACCESS REPORTS

Access 97 Report is designed for sorting, formatting, summarizing selected data, and others. Access reports can be exported to Microsoft Word. IRSR tracking reports have been sorted by KTI, PMR, and Key Attributes of System. Due to the complex nature of the IRSR tracking database, the IRSR tracking report may consist of several subreports with a total page count of 150 or more. Therefore, no IRSR tracking report is attached to this document.

D.2.3 LINK TO OTHER ACCESS DATABASES

The IRSR Tracking Database is also designed to link with the FEP Database and IRSR GODO Database. The FEP number and IRSR number have been included in the IRSR tracking database. The IRSR Tracking Database can relate to IRSR Acceptance Criteria table for the detail acceptance criteria as well as to FEP Database for specific FEP information.

D.3 WORKING WITH ISSUE RESOLUTION STATUS REPORTS TRACKING DATABASE

The IRSR Tracking Database provides a useful tool to assess how PMRs address IRSR acceptance criteria. A switchboard is built to give the user direct access to the common analysis, such as KTI subissues and process model factors analysis and analysis of acceptance criteria that have not been addressed in the database. The results of both analyses are discussed below.

D.3.1 KEY TECHNICAL ISSUES SUBISSUES AND PROCESS MODEL FACTORS ANALYSIS

In the IRSR tracking database, process model factors are assigned to KTI subissues creating a link from KTI subissues, process model factors, to FEPs. The summary of the KTI subissues addressed by PMR(s) and process model factors(s) is presented in Table D.3-1. It should be mentioned that the subissue title in each IRSR is slightly different from Appendix B, List of Subissues in NRC Key Technical Issues, in TSPA&I IRSR Rev 2 (NRC 2000 [149372]).

Table D.3-1. Summary of Process Model Reports and Process Model Factors that Address Nine Issue Resolution Status Report Key Technical Issues

KTI Subissue	PMR	RSS4/Process Model Factor
Container Lifetime and Source Term (CLST) IRSR		
Acceptance criteria applicable to all six subissues	DE[1]	All DE process model factors
	EBS[2]	All EBS process model factors
	Waste Form[3]	All WF process model factors
	Waste Package[4]	All WP process model factors
CLST 1: Effects of Corrosion Processes on Container Lifetime	EBS[2]	In-Drift Moisture Distribution
	Waste Package[4]	Waste Package Degradation and Performance
CLST 2: Effects of Phase Instability and Initial Defects on Mechanical Failure and Container Lifetime	Disruptive Events[1]	All DE process model factors
	Waste Package[4]	Waste Package Degradation and Performance
CLST 3: Rate of SNF Radionuclide Release from EBS	EBS[2]	In-Drift Physical and Chemical Environment
	Waste Form[3]	Commercial Spent Nuclear Fuel Degradation and Performance
CLST 4: Rate of HLW Glass Radionuclide Release from EBS	EBS[2]	In-Drift Moisture Distribution
	Waste Form[3]	Defense High-Level Radioactive Waste Degradation and Performance
CLST 5: Effects of In-Package Criticality on WP and EBS Performance	(The subissue of criticality is addressed in a topical report on nuclear criticality and supporting documents.)	
CLST 6: Effects of Alternative EBS Designs on Container Lifetime and Radionuclide Release	Disruptive Events[1]	All DE process model factors
	EBS[2]	In-Drift Physical and Chemical Environment
	Waste Package[4]	Waste Package Degradation and Performance
Evolution of the Near-Field Environment (ENFE) IRSR		
ENFE 1: The Effects of Coupled Thermal-Hydrologic-Chemical Processes on Seepage and Flow	EBS[2]	In-Drift Moisture Distribution
	Near-Field Environment[5]	Coupled Effects on Seepage
	UZ Flow and Transport[6]	Coupled Effects on UZ Flow
ENFE 2: Effects of Coupled Thermal-Hydrologic-Chemical Processes on the Waste Package Chemical Environment	EBS[2]	In-Drift Physical and Chemical Environment
	Near-Field Environment[5]	Coupled Effects on Seepage
ENFE 3: The Effects of Coupled Thermal-Hydrologic-Chemical Processes on the Chemical Environment For Radionuclide Release	EBS[2]	In-Drift Physical and Chemical Environment
	Near-Field Environment[5]	Coupled Effects on UZ Radionuclide Transport
	Waste Form[3]	In-Package Environments

Table D.3-1. Summary of Process Model Reports and Process Model Factors that Address Nine Issue Resolution Status Report Key Technical Issues (Continued)

KTI Subissue	PMR	RSS4/Process Model Factor
ENFE 4: The Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers	EBS[2]	In-Drift Physical and Chemical Environment
	Near-Field Environment[5]	Coupled Effects on UZ Radionuclide Transport
	UZ Flow and Transport[6]	Coupled Effects on UZ Flow
ENFE 5: Coupled Thermal-Hydrologic-Chemical Processes Affecting Potential Nuclear Criticality in the Near Field	(The subissue of criticality is addressed in a topical report on nuclear criticality and supporting documents.)	
Igneous Activity (IA) IRSR		
IA 1: Probability of Future Igneous Activity	Disruptive Events[1]	Igneous Activity
IA 2: Consequences of Igneous Activity Within the Repository Setting	Biosphere[7]	Biosphere Dose Conversion Factors
	Disruptive Events[1]	Igneous Activity
Radionuclide Transport (RT) IRSR		
RT 1: Radionuclide Transport Through Porous Rock	SZ Flow and Transport[8]	SZ Radionuclide Transport
	UZ Flow and Transport[6]	Coupled Effects on UZ Flow
RT 2: Radionuclide Transport Through Alluvium	SZ Flow and Transport[8]	SZ Radionuclide Transport
RT 3: Radionuclide Transport Through Fractured Rock	SZ Flow and Transport[8]	SZ Radionuclide Transport
	UZ Flow and Transport[6]	Coupled Effects on UZ Flow
RT 4: Nuclear Criticality in the Far Field	(This subissue of criticality is addressed in topical report on nuclear criticality and supporting documents.)	
Repository Design and Thermal Mechanical Effects (RDTME) IRSR		
RDTME 1: Implementation of an Effective Design Control Process within the Overall Quality Assurance Program	(Criteria for this subissue are elements of the Design Control part of the QA program addressed in Chapter 3 of the QARD [DOE 2000 [149540]).	
RDTME 2: Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption	(This subissue will be addressed in Seismic Topical Report.)	
RDTME 3: Thermal-Mechanical Effects on Underground Facility Design and Performance	EBS[2]	In-Drift Physical and Chemical Environment
	Near-Field Environment[5]	Coupled Effects on UZ Flow

Table D.3-1. Summary of Process Model Reports and Process Model Factors that Address Nine Issue Resolution Status Report Key Technical Issues (Continued)

KTI Subissue	PMR	RSS4/Process Model Factor
Structural Deformation and Seismicity (SDS) IRSR		
SDS 1: Faulting	Disruptive Events[1]	Seismic Activity
SDS 2: Seismicity	Disruptive Events[1]	Seismic Activity
SDS 3: Fracturing and Structural Framework of the Geologic Setting	SZ Flow and Transport[8]	SZ Flow
	UZ Flow and Transport[6]	UZ Flow
SDS 4: Tectonic Framework of the Geologic Setting	Disruptive Events[1]	Igneous Activity
Thermal Effects on Flow (TEF) IRSR		
TEF 1: Sufficiency of thermal-hydrologic testing program to assess thermal reflux in the near field	EBS[2]	In-Drift Physical and Chemical Environment
	Near-Field Environment[5]	All NFE process model factors
TEF 2: Sufficiency of thermal-hydrologic modeling to predict the nature and bounds of thermal effects on flow in the near field	EBS[2]	In-Drift Physical and Chemical Environment
	Near-Field Environment[5]	Coupled Effects on Seepage
	UZ Flow and Transport[6]	Coupled Effects on UZ Flow
TEF 3: Adequacy of total system performance assessment with respect to thermal effects on flow	EBS[2]	In-Drift Physical and Chemical Environment
	UZ Flow and Transport[6]	Coupled Effects on UZ Flow
Unsaturated Zone (UZ) and Saturated Zone (SZ) Flow Under Isothermal Conditions (USFUIC) IRSR		
USFIC1: Climate Change	UZ Flow and Transport[6]	Climate
USFIC2: Hydrologic Effects of Climate Change	SZ Flow and Transport[8]	SZ Flow
	UZ Flow and Transport[6]	Climate
USFIC3: Present-Day Shallow Infiltration	UZ Flow and Transport[6]	Net Infiltration
USFIC4: Deep Percolation (Present and Future [Post-Thermal Record])	UZ Flow and Transport[6]	Seepage into Emplacement Drifts
USFIC5: Saturated Zone Ambient Flow Conditions and Dilution Processes	SZ Flow and Transport[8]	SZ Flow
USFIC6: Matrix Diffusion	SZ Flow and Transport[8]	SZ RT
	UZ Flow and Transport[6]	Coupled Effects on Seepage

Table D.3-1. Summary of Process Model Reports and Process Model Factors that Address Nine Issue Resolution Status Report Key Technical Issues (Continued)

KTI Subissue	PMR	RSS4/Process Model Factor
Total System Performance Assessment and Integration (TSPAI) IRSR, Rev 1		
TSPAI1: Demonstration of the overall performance objective (TSPAI4, in Revision 2)	(Criteria for this subissue will be established after the rule is published in final form.)	
TSPAI2: Demonstration of Multiple Barriers (TSPAI1, in Revision 2)	(Criteria for this subissue developed in Revision 2 of the IRSR were not available during the preparation of PMRs.)	
TSPAI3: Model Abstraction (TSPAI3, in Revision 2)	All	All
TSPAI4: Scenario analysis (TSPAI2, in Rev 2)	All	All

Source: [1]CRWMS M&O 2000 [141733], [2]CRWMS M&O 2000 [145796], [3]CRWMS M&O 2000 [138332], [4]CRWMS M&O 2000 [138396], [5]CRWMS M&O 2000 [146589], [6]CRWMS M&O 2000 [145774], [7]CRWMS M&O 2000 [151615], [8]CRWMS M&O 2000 [145738]

Because TSPA&I IRSR Rev 2 (NRC 2000 [149372]) was issued after the PMRs were prepared, the field of PMR Approach and Section Reference in the current IRSR tracking database addresses TSPA&I IRSR, Rev 1 (NRC 1999 [103760]).

D.3.2 ACCEPTANCE CRITERIA HAVE NOT BEEN ADDRESSED

Using a query to select the blank cells of PMR Approach and Section Reference field can screen out the acceptance criteria that have not been addressed by PMRs. The query result of the IRSR tracking database shows that all of the applicable acceptance criteria have been addressed by PMRs.

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APPENDIX E

ANALYSES MODEL AND DATA TRACEABILITY

APPENDIX E

ACRONYMS AND ABBREVIATIONS

AMR	Analysis Model Report
ADTN	automated data tracking number
CSNF	commercial spent nuclear fuel
DIRS	Document Input Reference System
DOE	U.S. Department of Energy
DSNF	DOE-owned spent nuclear fuel
DTN	data tracking number
EBS	engineered barrier system
FEHM	finite element heat and mass
HIC	hydrogen induced cracking
HLW	high-level radioactive waste
NFE	near-field environment
NQ	not qualified
PMR	Process Model Report
RIP	Repository Integrated Performance model
SCC	stress corrosion cracking
SZ	saturated zone
TH	thermal hydrologic
TSPA	total system performance assessment
UZ	unsaturated zone

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APPENDIX E

ANALYSES MODEL AND DATA TRACEABILITY

The traceability of information from the Total System Performance Assessment–Site Recommendation back to its supporting documents is presented in this appendix. Several different approaches may be used to illustrate the traceability of the technical information used as the basis for the TSPA-SR. The individual sections of Chapter 3 have detailed the major inputs and assumptions used in the TSPA-SR. In this attachment, the sources of information and the flow of information in the TSPA-SR model are summarized. Additional details of the technical basis for the information in the TSPA-SR model may be found in the TSPA-SR Model Document (CRWMS M&O 2000 [148384]).

The primary technical input from each of the individual component parts of the TSPA-SR are contained in a hierarchy of Analysis/Model Reports (AMRs). The output of the source AMRs was either a data set (represented by a Data Tracing Number [DTN]) or an analytical expression or model that is implemented in the construct of the TSPA-SR model. Table E-1 summarizes each of the AMRs and corresponding DTNs that provided primary source information to the TSPA-SR. The AMRs represented on this table correspond to the source AMR used as the input as identified on the Document Input Reference System (DIRS) for the TSPA-SR Model Document (CRWMS M&O 2000 [148384]). The DTNs identified on this table correspond to the source DTNs used as direct input to the TSPA-SR Model. Generally, the DTN is created in the corresponding AMR. However, in some cases as noted in the footnote to this table, the source DTN is not derived directly from the applicable reference document. In these cases, the reference document discusses the applicability and appropriateness of the cited DTN for the purposes of evaluating the long-term performance of the potential Yucca Mountain repository system.

Figure E-1 shows schematically the integration of the AMR feeds within the TSPA-SR model itself. Each of the colored boxes on the figure represents one or more AMRs that directly feeds the TSPA model. The legend identifies the Process Model Report (PMR) category that each AMR supports, with each color representing a different PMR.

Both Table E-1 and Figure E-1 depict the final analyses, model and data feeds into the TSPA-SR model. It is important to recognize that these final feeds are built on a hierarchy of supporting AMRs, calculations, data and software that are contained in a family of supporting AMRs and their associated DTNs. Figures E-2 through E-7 depict the entire hierarchy of AMRs and DTNs that form the entire scientific basis for the TSPA-SR. This volume of work is summarized in the PMRs, discussed in Section 3.1. As with Figure E-1, these lower level model integration figures were developed by tracing the data and information flow through the DIRS and the Automated Technical Data Tracking (ATDT) system. The information available in these document and data tracking systems as of August 31, 2000 was used as the basis for these figures. As the supporting analyses and models are changed or revised, the actual information cited as being directly relied upon will also change.

The figures and tables that are presented here have been referred to as the "family trees" with the Total System Performance Assessment (TSPA) Model diagram. The purpose of the "family trees" is to show the interrelationships and integration of the AMRs that were used as inputs or information sources in developing the TSPA model. These "family trees" provide a more complete indication of the scientific basis for the analyses, models and data used in the development of the TSPA-SR model. It is beyond the scope of this document to summarize all of this technical information. That is the role of the nine individual PMRs. However, providing these roadmaps through the information flow leading up to the TSPA-SR allows the reader to trace the source information as far back as desired.

It bears noting that the family of AMRs that directly or indirectly support the TSPA-SR indicated on Figures E-1 through E-7 does not include a number of AMRs used to support feature, event and process screening arguments. These AMRs are indicated separately in Appendix B.

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model

Key System Attribute	Factor	Reference Document	Major Types of Input Parameters to the TSPA-SR	Data Tracking Number
Limiting Water contacting waste package	Climate	*Future Climate Analysis* (USGS 2000 [136368])	Climate states Timing and sequence	GS000308315121.003 [151139]
	Infiltration	*Analysis of Infiltration Uncertainty* (CRWMS M&O 2000 [143244])	Probabilities for different infiltration scenarios	SN0003T0503100.001 [149556]
	UZ flow above potential repository	*Abstraction of Flow Fields for RIP* (CRWMS M&O 2000 [123913])	Flow fields for different infiltration scenarios and climate states	SN9910T0581699.002 [126110]
	Seepage into drifts	*Abstraction of Drift Seepage* (CRWMS M&O 2000 [142004])	Seepage flux and seepage fraction as a function of percolation flux	SN9912T0511599.002 [146902]
		Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux (CRWMS M&O 2000 [152204])	Percolation flux – f (multiple locations, waste type, time, climate)	SN0001T0872799.006 [147198]
	Coupled processes - effects on UZ flow	*Drift Scale Coupled Processes (DST and THC Seepage) Models* (CRWMS M&O 2000 [142022])	Flow fields affected by TH	N/A – Background Information Only
	Coupled processes - effects on seepage	*Abstraction of Drift Seepage* (CRWMS M&O 2000 [142004])	Seepage flux and seepage fraction as a function of percolation flux	SN9912T0511599.002 [146902]
		Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux (CRWMS M&O 2000 [152204])	Percolation flux – f (multiple locations, waste type, time, climate)	SN0001T0872799.006 [147198]
	In-drift physical and chemical environments	*Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	Temperature and relative humidity on the drip shield surface – f (multiple locations, waste type, time, climate)	SN0001T0872799.006 [147198]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Limiting Water contacting waste package (Continued)	In-drift physical and chemical environments (Continued)	*In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818])	pH – f (region, time), response surface Chloride – f (region, time)	MO0002SPALOO46.010 [149168]
	In-drift moisture distribution	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	Seepage flux through the drip shield Fraction of drip shield surface that is wet	N/A – Equations
	Performance of drip shield	*Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier* (CRWMS M&O 2000 [146460])	Threshold for general corrosion initiation	LL991212305924.108 [144927]
		Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis (CRWMS M&O 2000 [147641])	General corrosion rate under drip and no-drip conditions	MO0003SPASUP02.003 [147299]
		WAPDEG Analysis of Waste Package and Drip Shield Degradation (CRWMS M&O 2000 [146427])	Drip shield geometry and thickness Drip shield failure time history Number of penetration openings in drip shield by general corrosion, crevice corrosion, SCC, HIC, and other degradation modes	N/A – Design information and WAPDEG Outputs
Long Waste package lifetime	Moisture, temperature, and chemistry effects on waste package	*In-Drift Precipitates/Salts Analysis* (CRWMS M&O 2000 [127818])	pH – f (region, time, response surface) Chloride – f (region, time, response surface) Ionic Strength	MO0002SPALOO46.010 [149168]
		Draft of AMR *Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	Average and maximum temperature on waste package surface – f (waste type, region, time, climate) Temperature and relative humidity on waste package surface – f (multiple locations, waste type, time, climate, infiltration)	SN0001T0872799.006 [147198]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Long Waste package lifetime (Continued)	Moisture, temperature, and chemistry effects on waste package (Continued)	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	Seepage flux through waste package Fraction of waste package surface that is wet	N/A – Equations
	Performance of waste package barrier	*WAPDEG Analysis of Waste Package and Drip Shield Degradation* (CRWMS M&O 2000 [146427])	Waste package geometry Thickness of waste package barriers Waste package failure time history Number of penetration openings in waste package by general corrosion, crevice corrosion, SCC, and other degradation modes	N/A – Design Input and WAPDEG Output
		Aging and Phase Stability of Waste Package outer Barrier (CRWMS M&O 2000 [147639])	Kinetics of secondary phase formation in base metal and weld of waste package outer barrier Threshold secondary phase volume fraction above which corrosion resistance of waste package outer barrier is affected	N/A – Equations
		Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier (CRWMS M&O 2000 [146460])	Threshold relative humidity for general corrosion initiation under drip (after drip shield failure) and no-drip conditions	LL991212305924.108 [144927]
		General Corrosion and Localized Corrosion of Waste Package Outer Barrier (CRWMS M&O 2000 [144229])	General corrosion rate under drip (after drip shield failure) and no-drip conditions	LL000112205924.112 [141284]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Long Waste package lifetime (Continued)	Performance of waste package barrier (Continued)	*Abstraction of Models for Pitting and Crevice Corrosion of Drip Shield and Waste Package Outer Barrier* (CRWMS M&O 2000 [147648]) *Abstraction of Models of Stress Corrosion Cracking of Drip Shield and Waste Package Outer Barrier and Hydrogen Induced Corrosion of Drip Shield* (CRWMS M&O 2000 [135773])	Crevice corrosion initiation threshold of waste package outer barrier Pit density (under crevice) Pit penetration rate (under crevice) Stress and stress intensity factor profile in waste package outer barrier SCC initiation threshold SCC crack density SCC crack growth rate Effect of material and manufacturing defects on SCC initiation and crack growth rate SCC crack penetration opening size Effect of phase stability and aging of waste package outer barrier on SCC initiation and crack growth rate	MO0003SPAPCC03.004 [148992]
		Calculation of the Probability and Size of defect flaws in the Waste Package closure welds to support WAPDEG Analysis (CRWMS M&O 2000 [144551])	Probability of the occurrence of material and manufacturing defects Size of material and manufacturing defects	MO0001SPASUP03.001 [144567]
Limiting Radionuclide mobilization and release from the engineered barrier system	Moisture, temperature, and chemistry effects within waste package	*In-Package Chemistry Abstraction* (CRWMS M&O 2000 [129287])	pH – f (region, time) Total dissolved carbonate (CO_3^2) – f (region, time) Oxygen fugacity – f (region, time) Ionic strength – f (region, time) Fluoride – f (region, time) CO_2 fugacity	MO9911SPACDP37.001 [139569]
		Draft of AMR In-Package Source Term Abstraction (CRWMS M&O 2000 [152213]	Volume of water in the waste package/waste form cell, calculated internal to GoldSim	N/A – calculated internal to GoldSim

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Limiting Radionuclide mobilization and release from the EBS (Continued)	Commercial Spent Nuclear Fuel (CSNF) waste form (with cladding or canister) performance	*Inventory Abstraction* (CRWMS M&O 2000 [136383])	Number of packages Zircaloy-clad fuel Stainless steel-clad fuel Inventory per package (actual and adjusted for ingrowth) Mass fraction	SN0003T0810599.010 [151021]
		Clad Degradation – Summary and Abstraction (CRWMS M&O 2000 [147210])	Fraction of surface area of Zircaloy-clad CSNF exposed as a function of time	MO0004SPACLD07.043 [151368]
		CSNF Waste Form Degradation: Summary Abstraction (CRWMS M&O 2000 [136060])	CSNF intrinsic dissolution rate equation	N/A - Equation
	DOE Spent Nuclear Fuel (DSNF) and plutonium disposition waste form performance	*Inventory Abstraction* (CRWMS M&O 2000 [136383])	Number of packages Inventory per package (actual and adjusted for ingrowth) Mass fraction	SN0003T0810599.010 [151021]
		DSNF and Other Waste Form Degradation Abstraction (CRWMS M&O 2000 [144164])	DSNF constant dissolution rate DSNF fuel surface area	MO0003RIB00083.000 [150886]
	High Level Waste (HLW) glass waste form (including canister) performance	*Inventory Abstraction* (CRWMS M&O 2000 [136383])	Number of packages Inventory per package (actual and adjusted for ingrowth) Mass fraction	SN0003T0810599.010 [151021]
		Defense High Level Waste Glass Degradation (CRWMS M&O 2000 [143420])	HLW intrinsic dissolution rate equation Specific surface area	MO0007RIB00091.000 [151712]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Limiting Radionuclide mobilization and release from the EBS (Continued)	Dissolved radionuclide concentration limits	*Summary of Dissolved Concentration Limits* (CRWMS M&O 2000 [143569])	Concentration limits (solubilities) for all isotopes included in TSPA	MO0004SPASOL10.002 [151713]
	Colloid-associated radionuclide concentrations	*Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary* (CRWMS M&O 2000 [125156])	Types of waste form colloids Concentration of colloids K_d and/or K_c for various colloid types Fraction of inventory that travels as irreversibly attached to colloids	MO0003SPAHIG12.002 [147949] MO0003SPAHLO12.004 [147952] MO0003SPAION02.003 [147951] MO0003SPALOW12.001 [147953] *LL991109851021.095 [142902] *MO0004SPAKDS42.005
	EBS radionuclide migration—transport through invert	*Draft of AMR Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux* (CRWMS M&O 2000 [152204])	Thermally perturbed saturation in the invert – f (waste type, region, time, climate) Saturation in the invert after thermal pulse – f(time)	SN0001T0872799.006 [147198]
	EBS radionuclide migration—transport through invert (continued)	*EBS Radionuclide Transport Abstraction* (CRWMS M&O 2000 [129284])	Invert geometry Porosity of the invert Diffusion coefficient	*SN9908T0872799.004 [108437] *MO0002SPASDC00.002 [148338]
Slow Transport away from the EBS	UZ flow and transport—advective pathways	*Particle Tracking Model and Abstraction of Transport Processes* (CRWMS M&O 2000 [141418])	FEHM particle-tracking model coupled to GoldSim (LANL 1999 [146971]) Grid nodes for each bin	N/A – not data
		Unsaturated Zone and Saturated Zone Transport Properties (CRWMS M&O 2000 [141440])	Transport parameters Fracture aperture in different unitsDispersivity of fracturesDispersivity of matrix	LA0003AM831341.001 [148751] LA0003JC831362.001 [149557]
		Abstraction of Flow Fields for RIP (CRWMS M&O 2000 [123913])	Flow fields for different infiltration scenarios and climate states (FEHM input files for the particle-tracking model [LANL])	SN9910T0581699.002 [126110]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Slow Transport away from the EBS (Continued)	UZ flow and transport—sorption and matrix diffusion	*Unsaturated Zone and Saturated Zone Transport Properties* (CRWMS M&O 2000 [141440])	K_d for all isotopes included in TSPA Matrix diffusion coefficients – f (isotopes, units)	LA0003AM831341.001 [148751] (K_d) LA0003JC831362.001 [149557] (Matrix)
	UZ flow and transport—colloid-facilitated transport	*Unsaturated Zone Colloid Transport Model* (CRWMS M&O 2000 [122799])	K_c and/or kinetic colloid parameters for plutonium, americium, thorium, etc. Colloid filtration factor	*LA0003MCG12213.002 [147285]
	Coupled processes—effects on UZ transport	*Unsaturated Zone and Saturated Zone Transport Properties* (CRWMS M&O 2000 [141440])	K_ds – f (isotopes, rock type)	LA0003AM831341.001 [148751]
	SZ flow and transport—advective pathways	*Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA* (CRWMS M&O 2000 [139440])	Breakthrough curves – f (radionuclide, region) Input parameters for convolution code Climate change flux multiplication factor Transport parameters Dispersivity (longitudinal, horizontal transverse, and vertical transverse) Boundary definition of the alluvium K_d for all isotopes included in TSPA Matrix porosity Flowing-interval spacing Effective diffusion coefficient Flowing interval Bulk density Source region definition Horizontal anisotropy K_c and/or kinetic parameters for plutonium desorption	SN0004T0501600.005 [151515] SN0002T0571599.002 [146931] TBV – 4431 SN0004T0571599.004 [149254] SNT05070198001.001 [143657]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Slow Transport away from the EBS (Continued)	Wellhead dilution	*Groundwater Usage by the Proposed Farming Community* (CRWMS M&O 2000 [144056])	Annual groundwater use	MO0003SPASGU01.003 [151075]
	Biosphere transport and uptake	*Abstraction of BDCF Distributions for Irrigation Periods* (CRWMS M&O 2000 [144054])	Biosphere dose conversion factor – f (radionuclide, irrigation time)	MO0003SPAABS07.006 [148872]
		Distribution Fitting to the Stochastic BDCF Data (CRWMS M&O 2000 [144055])	Biosphere dose conversion factor – f (radionuclide, irrigation time)	MO0003SPAABS08.004 [148453]
Addressing effects of disruptive events	Intrusive indirect release	*Igneous Consequence Modeling for the TSPA-SR* (CRWMS M&O 2000 [139563])	Number of waste packages damaged by intrusion (for groundwater transport source term) In-drift chemical conditions	*SN0001T0801500.001 [147818] *MO9912SPAPAI29.002 [148596]
	Volcanic direct release	*Characterize Framework for Igneous Activity at Yucca Mountain, Nevada (T0015)* (CRWMS M&O 2000 [141044])	Annual probability of igneous intrusion into the waste	LA0004FP831811.004 [151391]
		Igneous Consequence Modeling for the TSPA-SR (CRWMS M&O 2000 [139563])	Input parameters for ASHPLUME Probability that an intrusion into the waste will result in one or more eruptive vents through the waste Number of vents through the waste for intrusions that result in one or more vents through the waste Wind direction factor	SN0006T0502900.002 [150856]
		Disruptive Event Biosphere Dose Conversion Factor Analysis (CRWMS M&O 2000 [143378])	Biosphere dose conversion factors – f (radionuclide)	MO0002SPADVE03.001 [150755]

Table E-1. General Listing Inputs to the Total System Performance Assessment-Site Recommendation Model (Continued)

Key System Attribute	Factor	Reference Document	Input Parameters to TSPA-SR	Data Tracking Number
Addressing Effects of disruptive events (Continued)	Volcanic direct release (Continued)	*Evaluate Soil/Radionuclide Removal by Erosion and Leaching* (CRWMS M&O 2000 [136281])	Factor to account for radionuclide removal from soil	SN9912T0512299.002 [136370]
	Seismic activity	*Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain, Nevada* (CRWMS M&O 1999 [130569])	Probability of seismicity and structural deformation	MO0004MWDRIFM3.002 [149092]

NOTES: N/A = not applicable; In some cases, indicated by an asterisk (*), the original source of the DTN is not the cited document. In these cases, the cited document is the source of the background information, and the DTN is the number which covers the input data in the ATDT.

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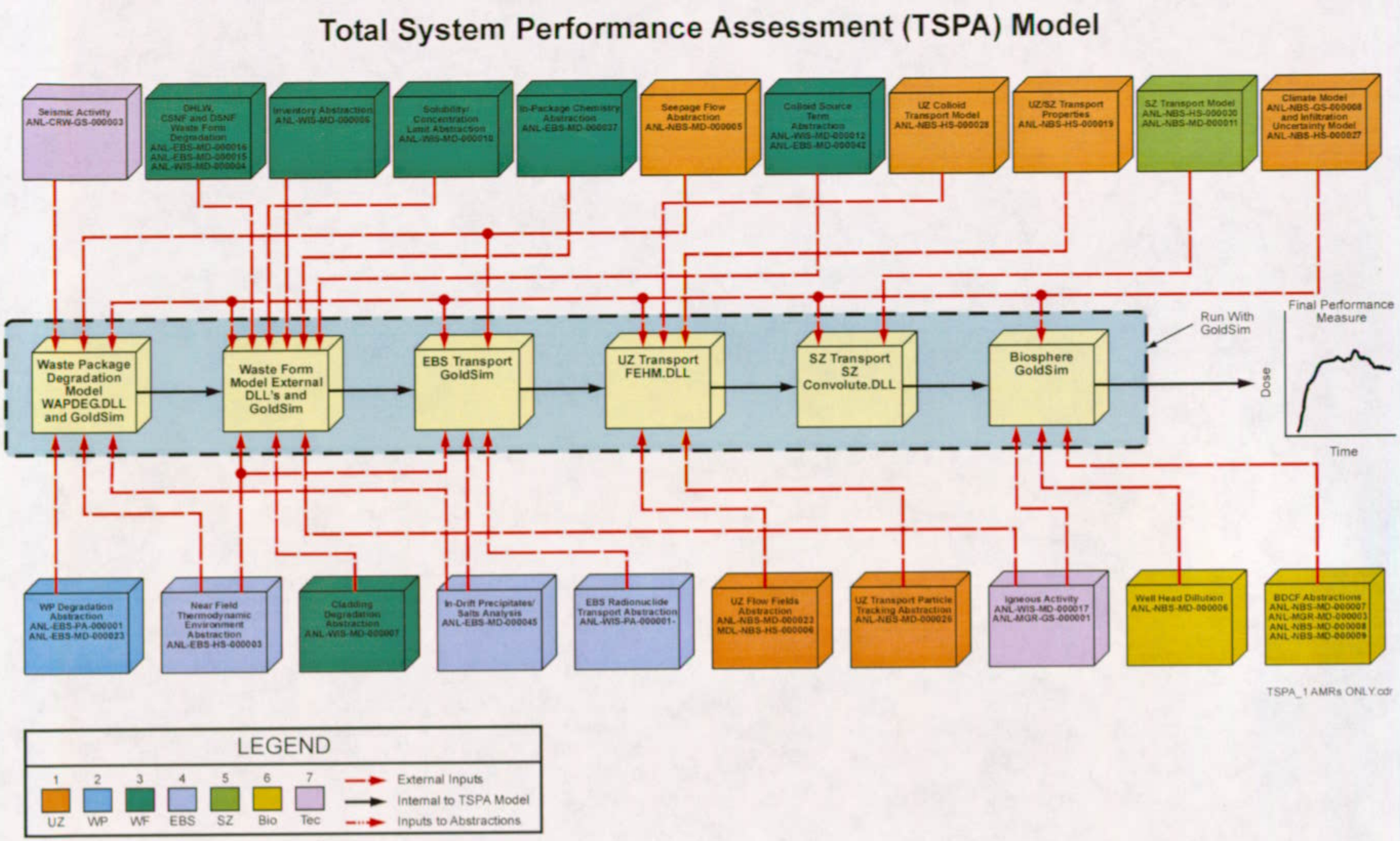
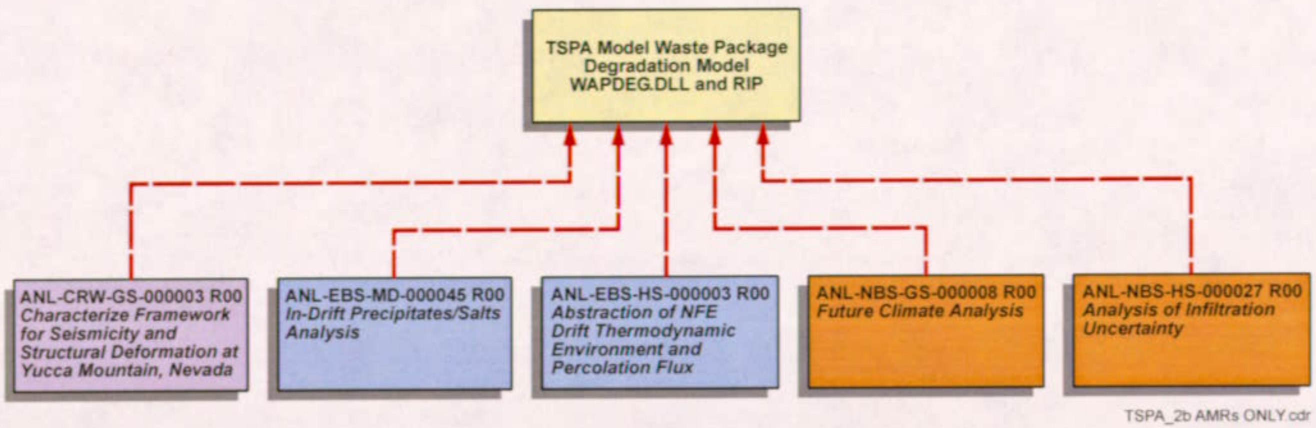


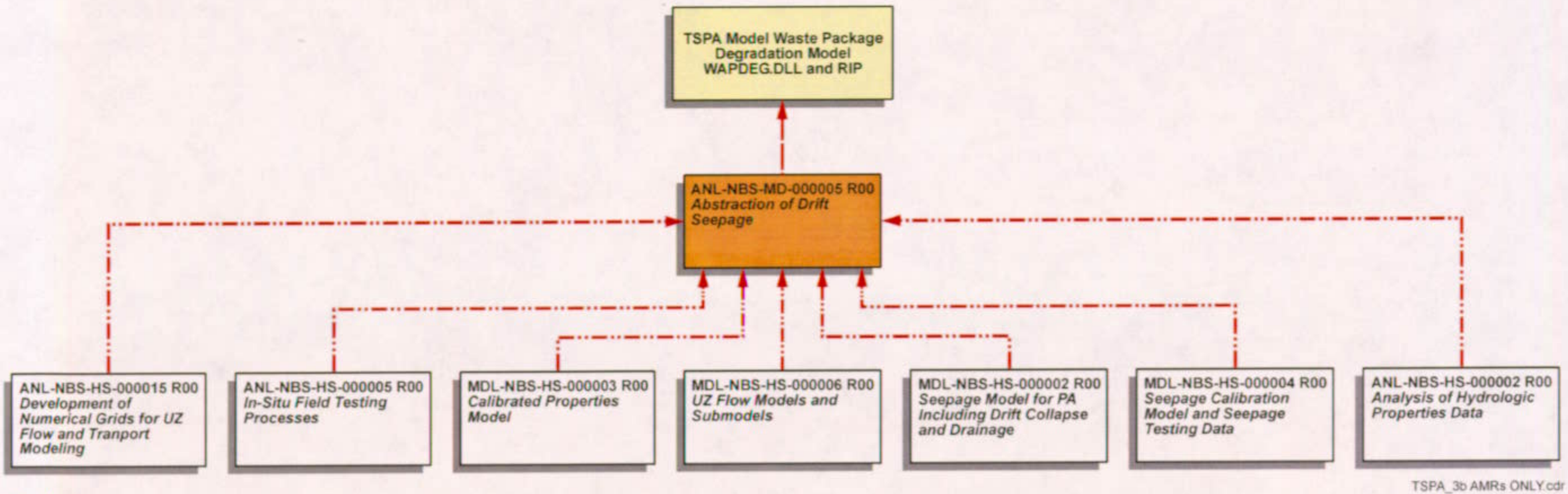
TSPA_1

Figure E-1. Hierarchy of Analyses and Models Supporting the Total System Performance Assessment-Site Recommendation



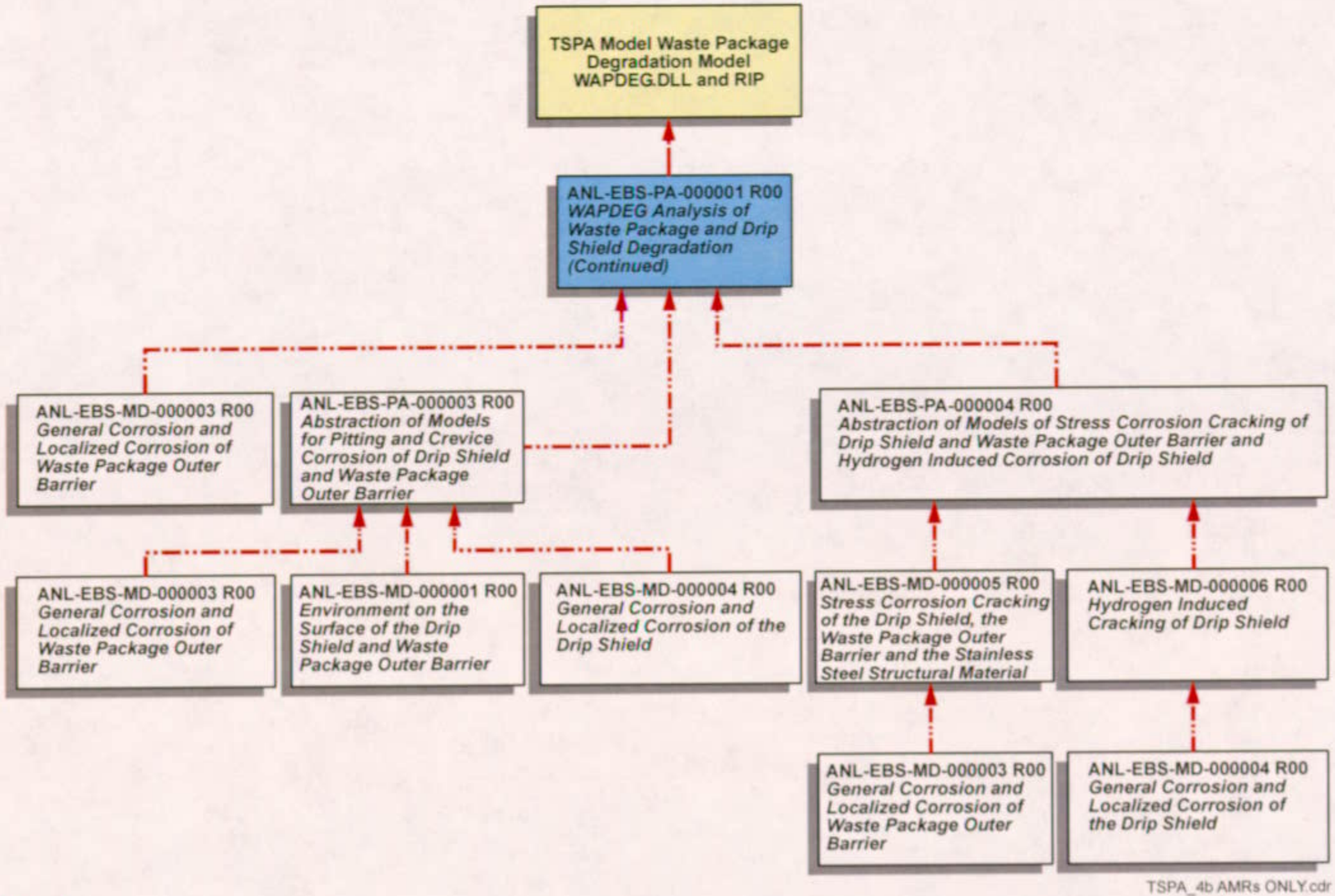
TSPA_2

Figure E-2a. Hierarchy of Analyses and Models Supporting the Waste Package Degradation Model of Total System Performance Assessment-Site Recommendation - Part A



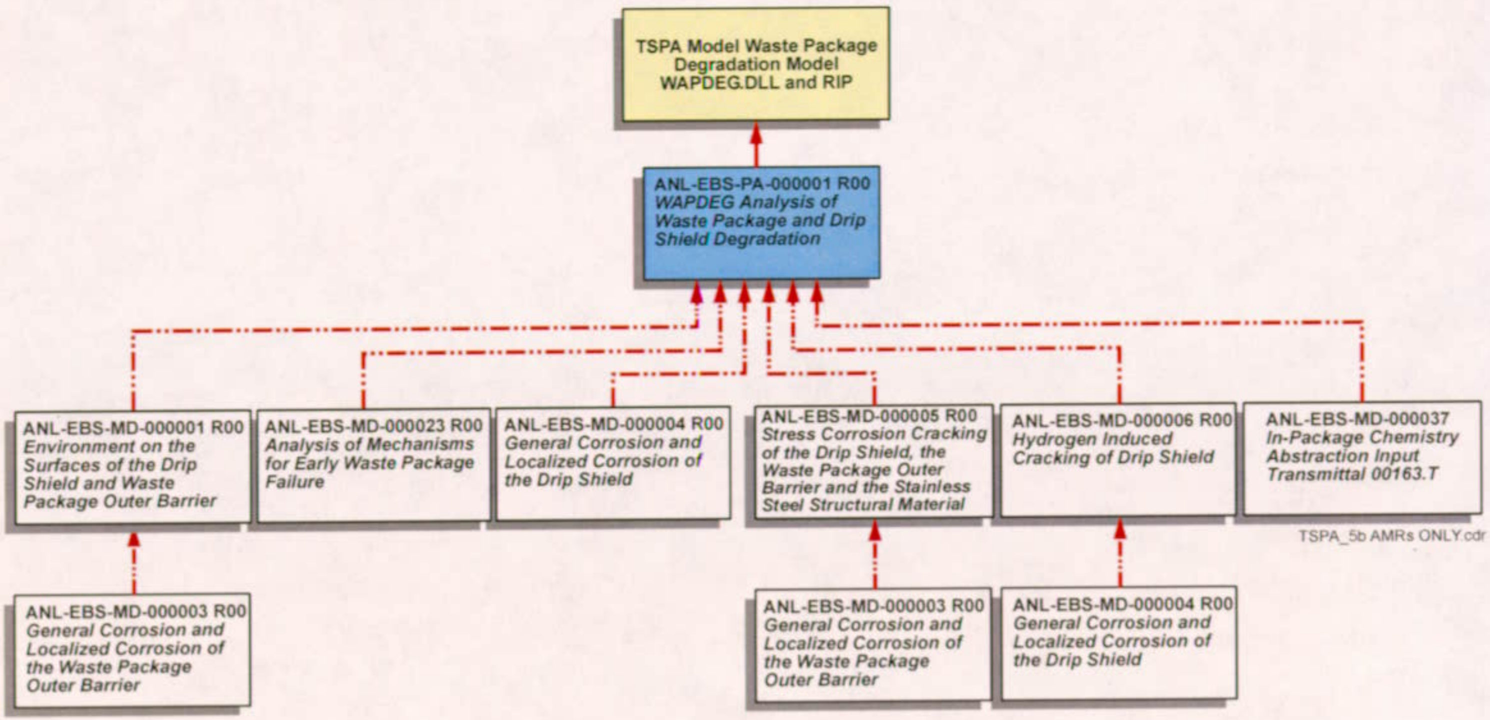
TSPA_3

Figure E-2b. Hierarchy of Analyses and Models Supporting the Waste Package Degradation Model of Total System Performance Assessment-Site Recommendation - Part B



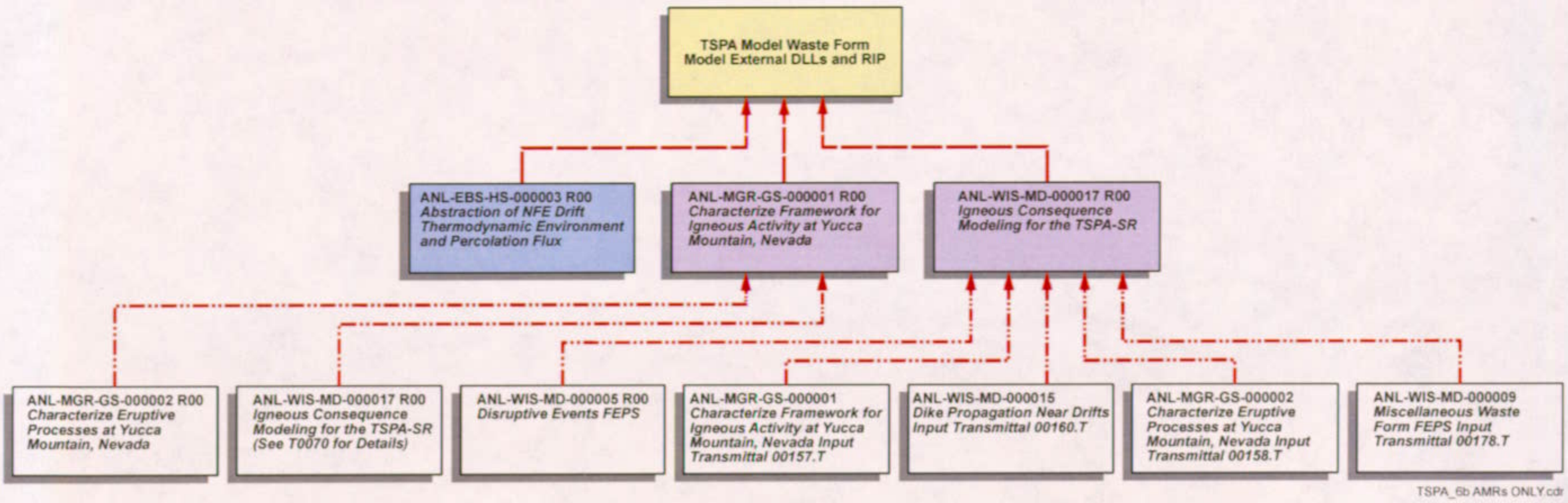
TSPA_4

Figure E-2c. Hierarchy of Analyses and Models Supporting the Waste Package Degradation Model of Total System Performance Assessment-Site Recommendation - Part C



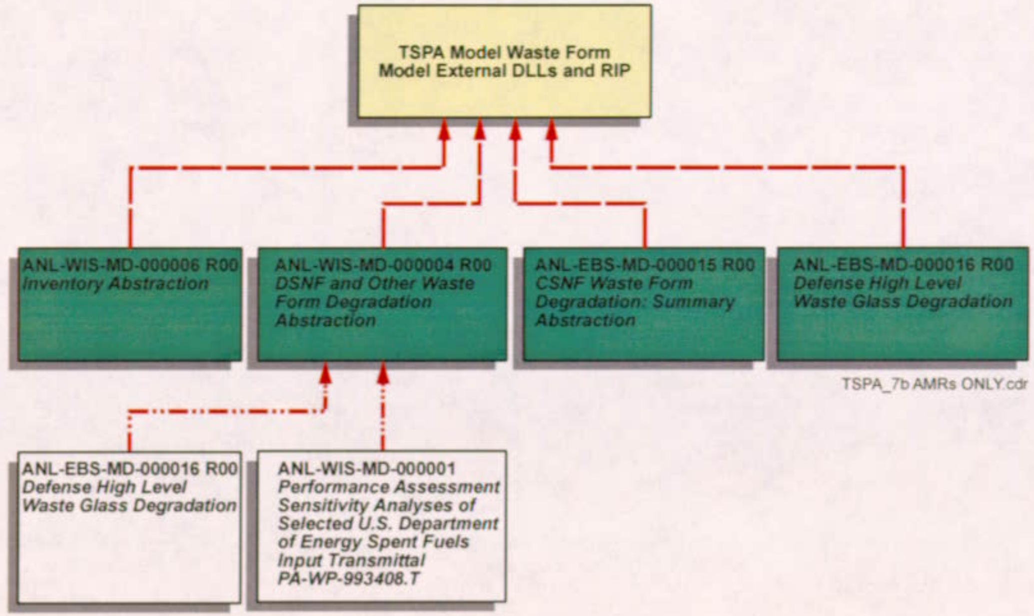
TSPA_5

Figure E-2d. Hierarchy of Analyses and Models Supporting the Waste Package Degradation Model of Total System Performance Assessment-Site Recommendation - Part D



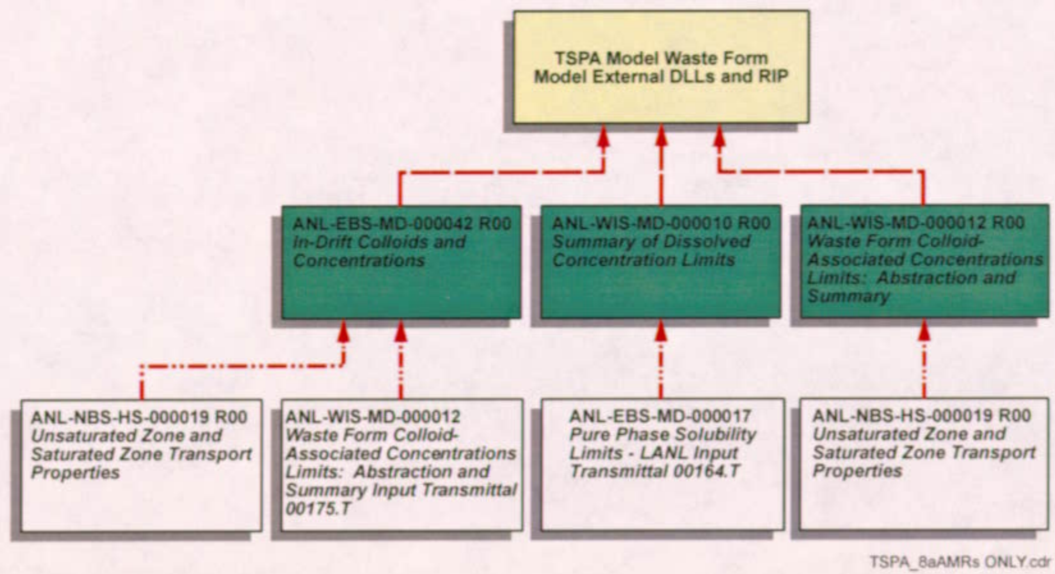
TSPA_6

Figure E-3a. Hierarchy of Analyses and Models Supporting the Waste Form Model of Total System Performance Assessment-Site Recommendation - Part A



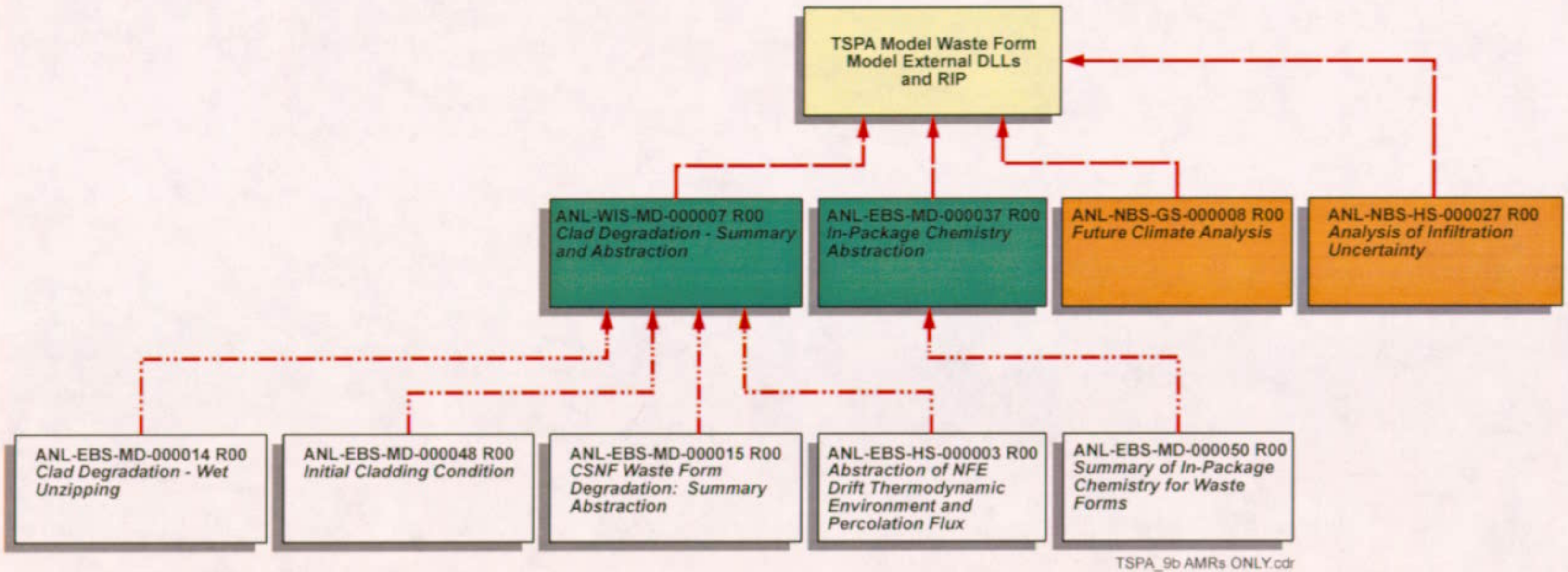
TSPA_7

Figure E-3b. Hierarchy of Analyses and Models Supporting the Waste Form Model of Total System Performance Assessment-Site Recommendation - Part B



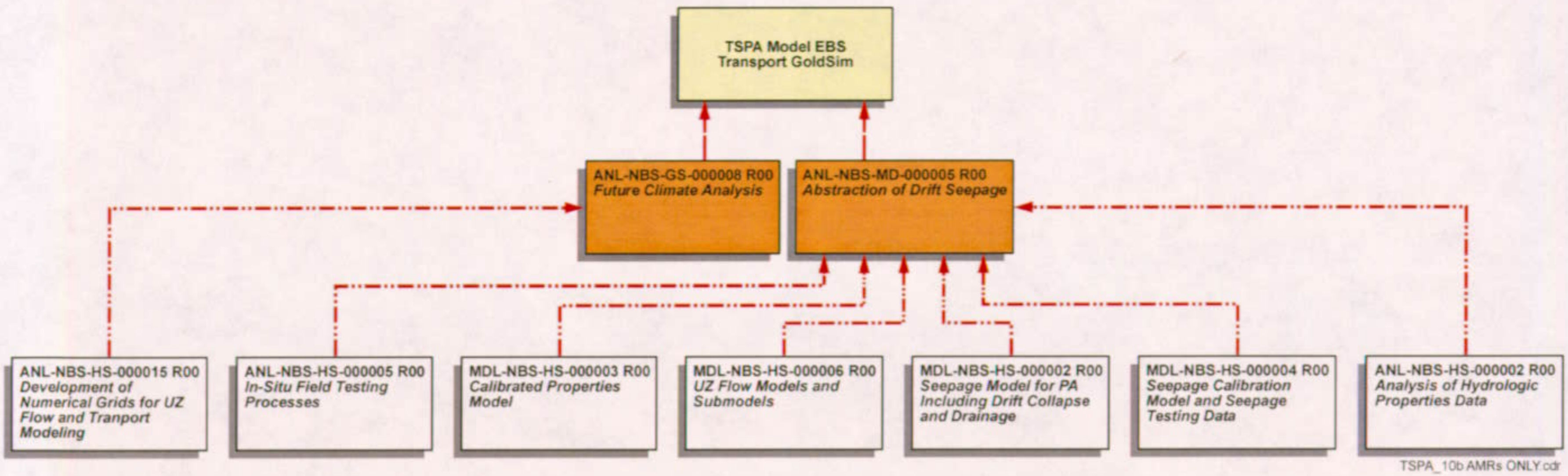
TSPA_8

Figure E-3c. Hierarchy of Analyses and Models Supporting the Waste Form Model of Total System Performance Assessment-Site Recommendation - Part C



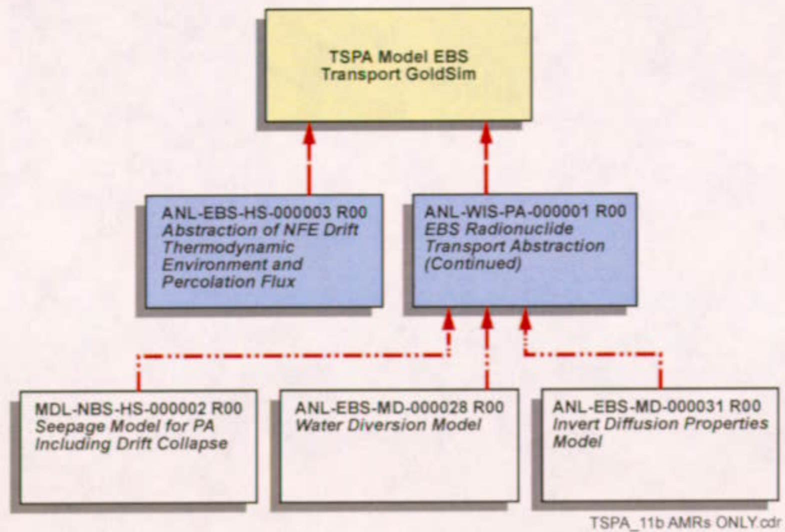
TSPA_9

Figure E-3d. Hierarchy of Analyses and Models Supporting the Waste Form Model of Total System Performance Assessment-Site Recommendation - Part D



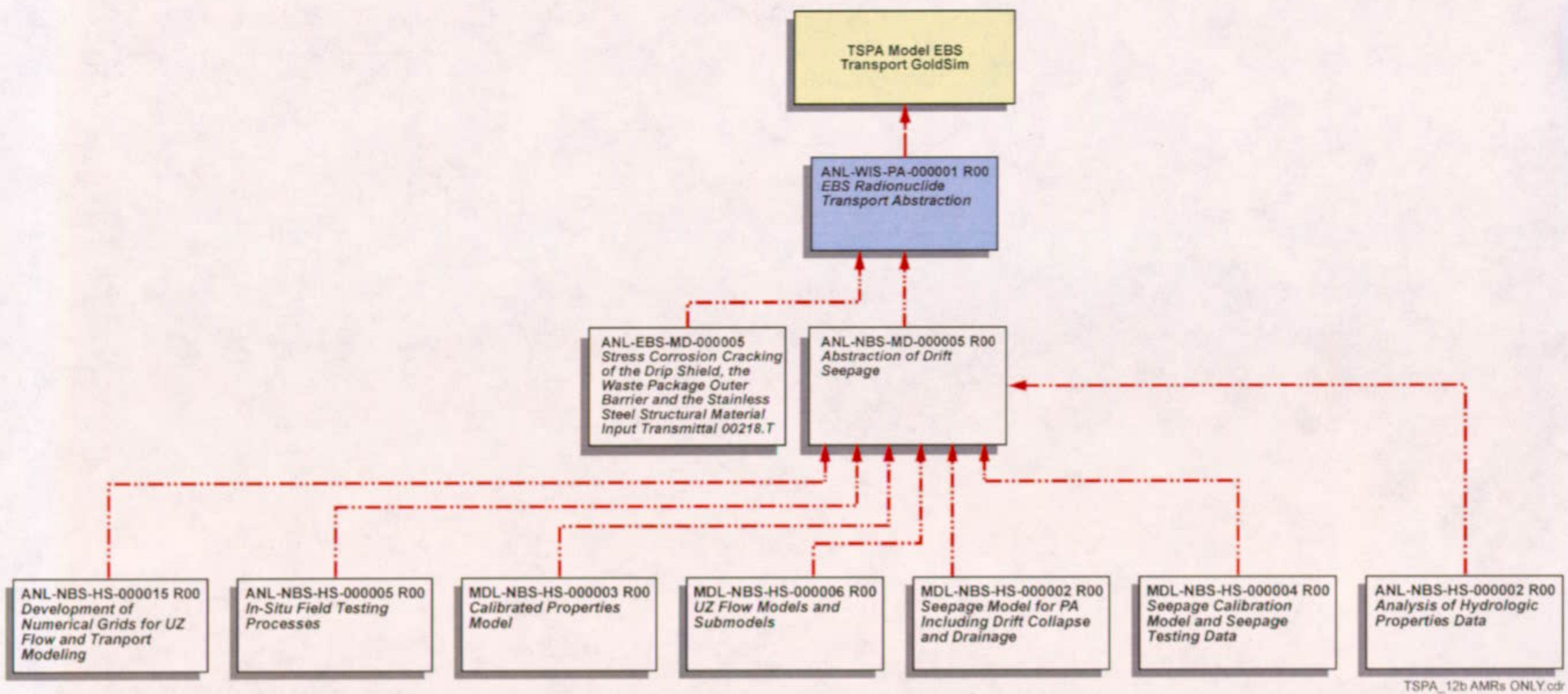
TSPA_10

Figure E-4a. Hierarchy of Analyses and Models Supporting the EBS Transport Model of Total System Performance Assessment-Site Recommendation - Part A



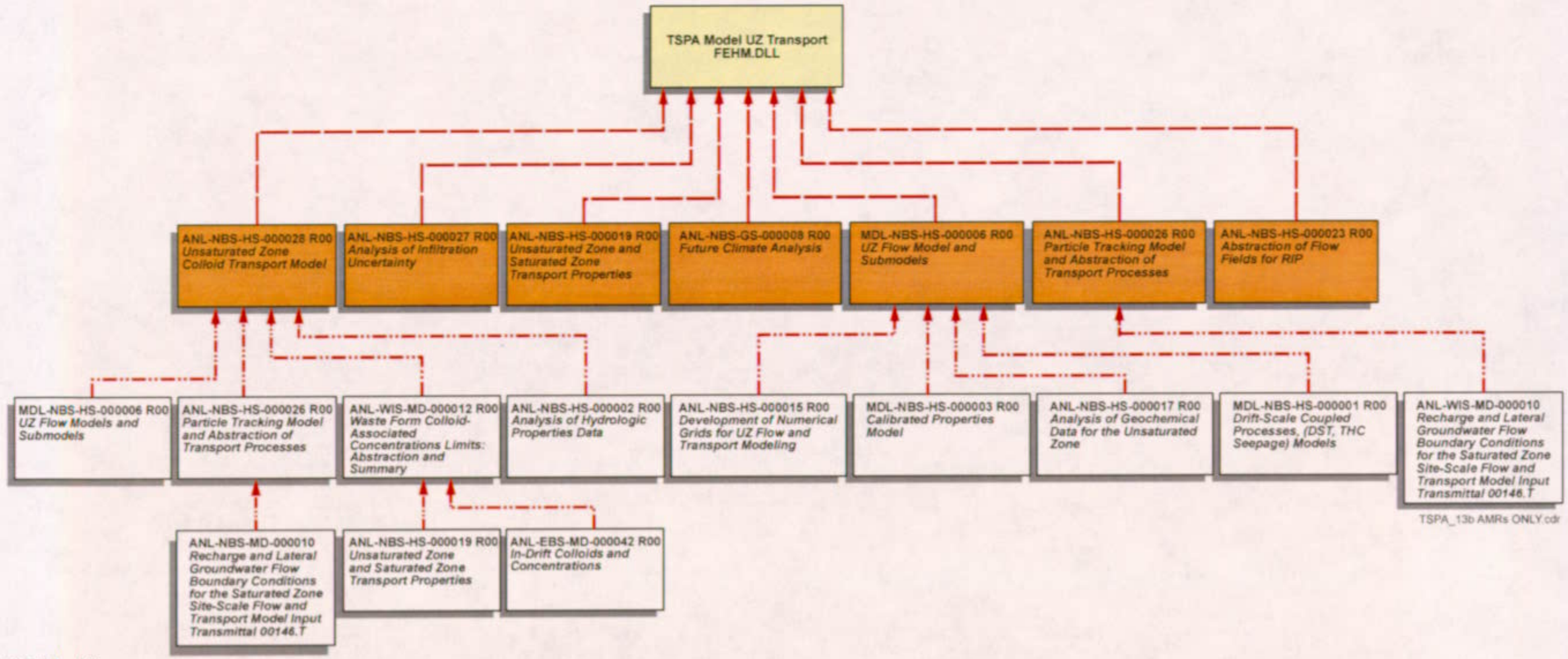
TSPA_11

Figure E-4b. Hierarchy of Analyses and Models Supporting the EBS Transport Model of Total System Performance Assessment-Site Recommendation - Part B



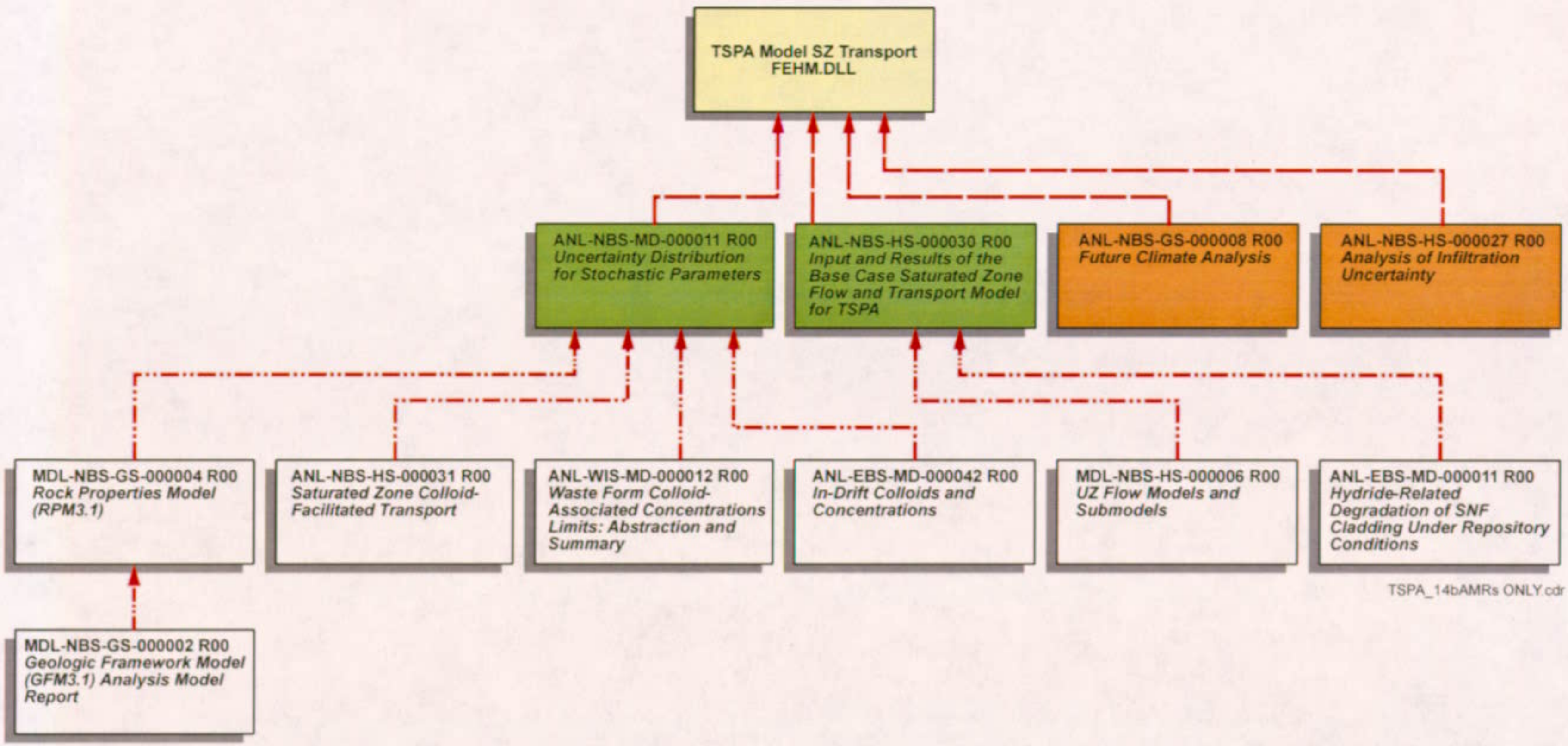
TSPA_12

Figure E-4c. Hierarchy of Analyses and Models Supporting the EBS Transport Model of Total System Performance Assessment-Site Recommendation - Part C



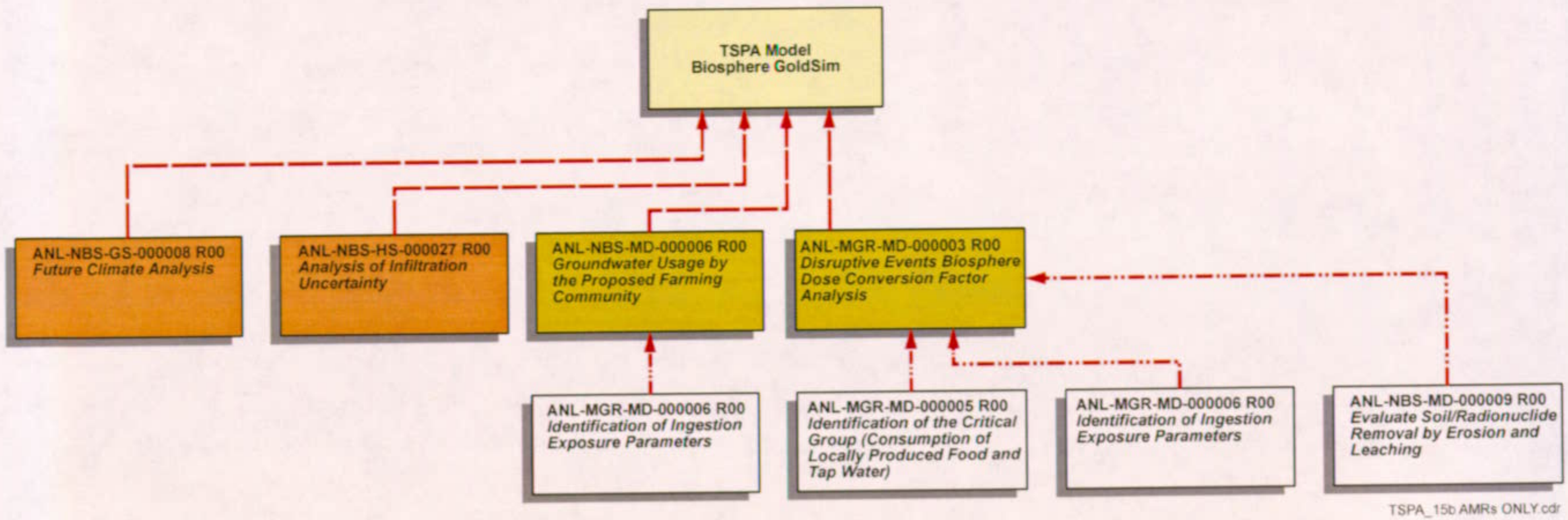
TSPA_13

Figure E-5. Hierarchy of Analyses and Models Supporting the Unsaturated Zone Transport Model of Total System Performance Assessment-Site Recommendation



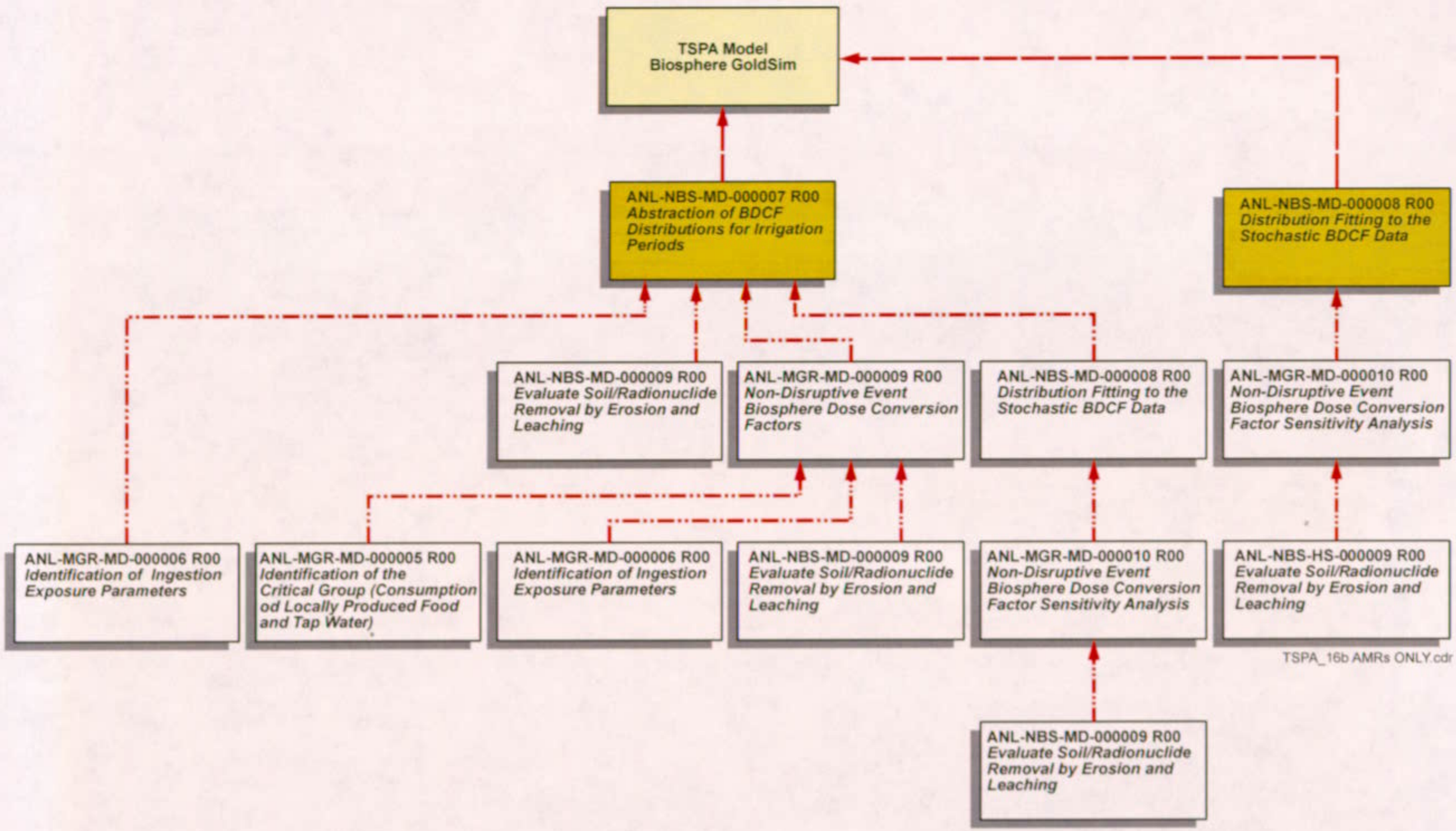
TSPA_14

Figure E-6. Hierarchy of Analyses and Models Supporting the Saturated Zone Transport Model of Total System Performance Assessment-Site Recommendation



TSPA_15

Figure E-7a. Hierarchy of Analyses and Models Supporting the Biosphere Model of Total System Performance Assessment-Site Recommendation - Part A



TSPA_16

Figure E-7b. Hierarchy of Analyses and Models Supporting the Biosphere Model of Total System Performance Assessment-Site Recommendation - Part B

APPENDIX F

SYNTHESIS OF MAJOR ASSUMPTIONS AND CONSERVATISMS INCLUDED IN TOTAL SYSTEM PERFORMANCE ASSESSMENT-SITE RECOMMENDATION

APPENDIX F

ACRONYMS AND ABBREVIATIONS

BDCF	Biosphere dose conversion factor
CSNF	Commercial spent nuclear fuel
DHLW	Defense high-level radioactive waste
DSNF	DOE-owned spent nuclear fuel
EBS	Engineered barrier system
HLW	high-level radioactive waste
NRC	U.S. Nuclear Regulatory Commission
RH	relative humidity
SZ	Saturated zone
SCC	stress corrosion crack
SNF	spent nuclear fuel
TSPA-SR	Total System Performance Assessment-Site Recommendation
UZ	Unsaturated zone
WP	waste package

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APPENDIX F

SYNTHESIS OF MAJOR ASSUMPTIONS AND CONSERVATISMS INCLUDED IN TOTAL SYSTEM PERFORMANCE ASSESSMENT-SITE RECOMMENDATION

Throughout Chapter 3, the major information feeds to the Total System Performance Assessment-Site Recommendation (TSPA-SR) have been summarized to provide the reader a complete picture of the key aspects of the individual component models of the TSPA-SR. This Appendix summarizes in one sense the major assumptions used in the individual component models and, as applicable, the degree of conservatism that has been incorporated in these models in order to reduce the need to explicitly quantify the uncertainty in each of the models.

This Appendix is broken into two parts. In the first discussion, the major assumptions and conservatisms associated with each component model of the TSPA-SR model are summarized (Tables F-1 to F-14). This summary is accomplished in the following sequence:

- Unsaturated zone (UZ) flow and transport
- Near field environment
- Engineered barrier system (EBS) chemical environment and radionuclide transport
- Drip shield/waste package (WP)
- Inventory
- In-package chemistry
- Commercial spent nuclear fuel (CSNF) degradation
- DOE-owned spent nuclear fuel (DSNF) degradation
- High-level radioactive waste (HLW) degradation
- Dissolved concentration limits
- Colloid concentration limits
- Saturated zone (SZ) flow and transport
- Biosphere
- Disruptive events.

The list of major assumptions focuses on those aspects within the TSPA-SR model that are conservative. "Conservative" as used here indicates that the assumptions may result in overestimation of the consequences of processes that have potential to degrade subsystem performance, or in the underestimation of processes that might result in improved subsystem performance. Because assumptions have been identified as being conservative based on subsystem-level analysis, rather than on the overall TSPA, they may not have a direct impact on the system-level performance. For example, assumptions that lead to overestimation of water flux through the drifts are identified as being conservative, but they will have no effect on overall performance as long as WP remain intact. Similarly, assumptions that tend to overestimate the annual dose rate due colloidal transport of radionuclides are listed as conservative even though the peak expected annual dose rate is dominated by transport of dissolved species that are unaffected by assumptions about colloids.

The root cause of most of the assumptions identified in Tables F-1 through F-14 relates to the fact that limited data exist to justify a more "realistic" representation of the particular process or component model. This is to be expected. As noted by U.S. Nuclear Regulatory Commission (NRC) and Environmental Protection Agency (EPA) in their rulemaking process, it is acknowledged that significant uncertainty about processes acting over the spatial (kilometers) and temporal (tens of thousands of years) scales of interest in post closure performance assessments will remain, even after a detailed site characterization program. That is, we are not seeking the exact answer, only an acceptably representative answer based on the information available. Where information is lacking or significant complexity exists, the approach taken in all component models of the TSPA has been to simplify and bound the representation. This approach has generally been followed in technical areas where either reducing the uncertainty is not feasible (i.e., additional data would not significantly reduce the uncertainty) or where the uncertainty does not significantly affect the performance assessment analyses (i.e., the TSPA results are insensitive to that particular component within the bounds of the expected models). Therefore, this lack of data identified in Tables F-1 through F-14 does not detract from the representativeness or appropriateness of the models developed to assess the post closure performance. However, it is useful to note these issues so the reader can best appreciate where and why certain assumptions have been made in the Analysis Model Reports that support the TSPA-SR.

Table F-1. Unsaturated Zone Flow and Transport

Description of Assumption	Basis for Assumption	Comments
Transport and flow through UZ.	Limited data.	Connectivity of fracture network likely overestimated. Residence time in perched water likely underestimated.
The treatment of fracture/matrix diffusion assumes a stagnant matrix.	This model was already implemented in finite element heat and mass and is very computationally efficient. It has been tested against analytical solutions for dual-porosity system, but it appears to be conservative for dual-permeability systems.	The use of a fracture/matrix diffusion model that allows matrix/matrix transport following diffusion has been shown to increase travel times in the UZ by tens of thousands of years for weakly sorbing radionuclides. We are currently implementing a new model that can allow this feature.
Colloid concentrations and K_ds used to calculate K_c are conservative (i.e., high).	There is so much uncertainty about colloid properties and transport that a more detailed representation was thought to be indefensible.	
Colloid retardation is neglected for all colloids in the UZ.	No data are available for R_c in the UZ or for reversible colloids.	
Physical colloid filtration is neglected for transport in fractures.	Lack of data to support including it.	
Diversion of flow & transport around the perched water rather than through it might be conservative.	We have no site information that can tell us the actual flow/transport paths in the vicinity of the perched water.	

Table F-2. Near-Field Environment

Description of Assumption	Basis for Assumption	Comments
Percolation flux 5 m above the drift crown is used as input to seepage calculations.	Seepage implemented conservatively into the model due to lack of a good basis for TM effects on seepage.	Water is always assumed to be present 5 meters above the crown of the drift so that there is always the possibility of seepage into the drift. Percolation flux is used as an input to the abstracted seepage model that governs the amount of the water entering the drifts.
Drift Seepage.	Limited data, especially for lower lithophysal unit; insufficient time (and data) to develop detailed model of thermal effects on seepage.	Thermal dry-out neglected, possible better performance of lower lithophysal unit neglected, seepage threshold possibly underestimated. Seep flux increased by 55 percent to account for effects of drift degradation & rock bolts. With present TSPA model seepage is not very important; this could possibly change if dependence of corrosion on seepage were put back in the model.

Table F-3. EBS – Chemical Environment And Radionuclide Transport

Description of Assumption	Basis for Assumption	Comments
All seepage into drifts is assumed to hit drip shields , not just seepage above drip shield (conservative).	Simplify modeling approach.	Taking credit for seepage locations requires a degree of precision not suitable for the seepage model, which was intended to define amount and probability of seepage.
Treatment of transport through WP and EBS.	Limited data.	Evaporation of water in WP is neglected. Radionuclide transport through a stress corrosion crack (SCC) is by diffusive transport through a thin, continuous film that is assumed to be always present. The advective flow for radionuclide transport through the invert is a one-dimensional process and always vertically downward. The binary diffusion coefficient for all radionuclides is bounded by the self-diffusion coefficient of water. A zero concentration boundary condition at the invert/UZ interface is enforced during the post-closure simulation period.
The potential beneficial effects from adsorption of radioisotopes on immobile colloidal material and other corrosion products are neglected.	Limited data.	This approach bounds uncertainties in adsorption behavior.
Colloid retardation by filtration in the invert is neglected.	Limited data.	Efforts at reduction in conservative treatment of colloid behavior are best spent on other areas.

Table F-4. Drip Shield/Waste Package

Description of Assumption	Basis for Assumption	Comments
Threshold relative humidity (RH) for corrosion initiation.	Insufficient information/data to quantify the corrosion initiation threshold RH for varying dripping conditions.	The current model uses a threshold RH based on the deliquescence point of NaNO3 salt, and the same threshold RH for both drip and no-drip conditions. Use of the same threshold RH for no-drip condition (including WP under non-failed drip shield) is conservative.
Density and orientation of embedded defects/flaws.	Lack of YMP-related data to quantify defect density and orientation.	The current model includes surface breaking defects and embedded defects in WP closure-lid welds, and assumes in the WP SCC analysis that all the defects are oriented in radial direction. Embedded defects/flaws are likely oriented parallel to the weld centerline and, if subject to SCC, will likely to grow circumferentially along the WP weld circumference, not radially as currently assumed in the WAPDEG nominal case model (CRWMS M&O 1998 [145618]).
SCC Crack Initiation and Propagation is assumed to occur under highly aggressive water chemistry conditions.	Limited data, both on corrosion and on environmental conditions in the potential repository.	The slip dissolution model used for SCC crack propagation in the WAPDEG analysis includes a term to account for effect of chemistry of water contacting the WP. The current WAPDEG analysis uses a range of values for the term that represent highly aggressive water conditions and materials much less resistant to SCC than Alloy 22.

Table F-5. Inventory Component

Description of Assumption	Basis for Assumption	Comments
C-14 included in inventory is transported entirely in the aqueous phase.	Simplify modeling approach.	This approach is conservative since less dilution occurs in aqueous transport. Also, gaseous transport of C-14 has been screened out conservatively assuming that all C-14 might be in the gas phase.

Table F-6. In-Package Chemistry Component

Description of Assumption	Basis for Assumption	Comments
Transport pathways inside container excluded; rather, container assumed to be a mixing cell.	Limited data and large uncertainty in physical environment in WP.	Inclusion of pathways would delay release of radioisotopes through the decrease in advective and diffusive transport. Would need to assume short-circuits to pathway similar to fracture flow in UZ.
Minimum pH in first 1000 yr. selected for each EQ3/6 (CRWMS M&O 1998 [102837]) run rather than using distribution.	Limited data and large uncertainty in chemical environment in WP.	Approach bounds uncertainty in low pH values and results in conservative radionuclide solubilities and waste-form degradation rates.

Table F-7. CSNF Degradation Component

Description of Assumption	Basis for Assumption	Comments
For CSNF degradation, no reducing conditions are assumed, which results in a fast degradation rate.	Limited data and large uncertainty in chemical environment in WP.	This fast rate is used to bound the upper end of the uncertainties.
Conservative estimate of perforation from creep rupture is used.	Limited data for non-irradiated cladding.	A conservative model for non-irradiated cladding was used that over estimates the creep of irradiated cladding by almost a factor of 30. Creep data by various European and American experimenters is available for irradiated cladding.
Unknown localized corrosion method for perforating cladding included.	Limited data and uncertainty in micro-chemical environment surrounding cladding.	In-package chemistry component calculates bulk water chemistry that can not account for microenvironments or transients of concentrating chemicals within the WP that could produce localized conditions for pitting corrosion of the cladding.
Wet unzipping has uncertainty added that is in addition to the uncertainty in the CSNF degradation rate; the unzipping propagates at between 1 and 240 times faster than the CSNF degradation rate.	Insufficient data and large uncertainty.	Wet unzipping has not been observed in spent fuel pools and might not occur at all because the phase transitions are in liquids and might not occur at the interface of secondary phases and cladding.

Table F-8. DSNF Degradation Component

Description of Assumption	Basis for Assumption	Comments
No cladding credit is assumed.	Simplify modeling approach, no technical basis for credit on DSNF.	This approach bounds DSNF degradation processes. The naval spent nuclear fuel (SNF) degradation model does account for the performance of the naval SNF cladding.
A constant degradation rate is used that conservatively bounds degradation of metallic uranium present in N-reactor fuel; degradation of DSNF can occur in one time step of TSPA.	Little quality assurance data exists on DSNF fuel and so a very conservative approach is used to bound uncertainty in the characteristics of the DSNF fuel.	Unlike some DSNF, naval SNF is completely characterized and the degradation of naval SNF was modeled using the same environmental conditions used for the degradation of commercial SNF.

Table F-9. High-Level Radioactive Waste Degradation Component

Description of Assumption	Basis for Assumption	Comments
High degradation rates are assumed.	Lack of data.	Using high degradation rate bounds uncertainties.
In addition to a high degradation rate, a very conservative cracking surface enhancement factor of 20 over the geometric surface area is used (this uncertainty plus the uncertainty in degradation rate could make the degradation rate 1000 times faster than observed in experiments).	Simplify modeling approach.	A very conservative enhancement factor of 20 is used to bound uncertainty in the true value.

Table F-10. Dissolved Concentration Component

Description of Assumption	Basis for Assumption	Comments
High Np solubility is predicted because Np_2O_5 is conservatively assumed to be the controlling solid phase.	Limited data.	Using high solubility bounds uncertainties. During the first 1000 years after breach when the pH is predicted to be low, Np solubility is between 10^{-4} and 10^{-1} M (similar to CRWMS M&O 1995 [100198]. After 1000 years, Np solubility is between 10^{-7} and 10^{-4} M (similar to TSPA-VA DOE 1998 [100550]), Volume 3.
Np or other radioisotopes are not incorporated into secondary phases of uranium are assumed.	The effects have been conservatively excluded.	Experimental data are lacking to confirm this phenomenon with sufficient reliability to be used in a reasonable assurance evaluation.

Table F-11. Colloidal Concentration Component

Description of Assumption	Basis for Assumption	Comments
Assumption that groundwater colloids consist of montmorillonite (smectite) clay minerals.	Limited data.	This assumption overestimates the mass of radionuclides attached to groundwater colloids.
Concentration of iron-(hydr)oxide corrosion-product colloids not linked to corrosion rates.	Limited data.	Concentration of corrosion-product colloids is calculated based on an ionic strength stability relationship; it is assumed that there is always a sufficient number of corrosion-product colloids to satisfy this relationship.
Stabilities of colloid types (waste form, groundwater, and corrosion-product colloids) are treated independently.	Limited data.	Colloid concentrations are determined by a stability relationship that is a function of solution ionic strength only. This relationship is applied separately to each colloid type as opposed to all colloids simultaneously and therefore tends to overestimate colloid concentrations.

Table F-12. Saturated Zone Flow and Transport

Description of Assumption	Basis for Assumption	Comments
Sorption Coefficients for Alluvium.	Uncertainty.	Uncertainty distributions for sorption coefficients in alluvial units used in TSPA-SR based on limited laboratory data. Sorption experiments for I sorption were incomplete; final values of K_d may be higher.
No Sorption in Fractures.	Uncertainty.	The radionuclide transport model in the SZ for TSPA-SR assumes (conservatively) no sorption onto fracture surfaces in the fractured units. Laboratory and field testing have not defensibly demonstrated this process.
Conceptual Model of Colloid Filtration.	Uncertainty and Modeling Restriction.	The conceptual model of colloid filtration for colloids with irreversibly attached radionuclides used in TSPA-SR assumes that colloids are retarded by filtration. Permanent removal of colloids by mechanical filtration is not considered in the SZ transport model.
K_c Model Parameters for Colloid-Facilitated Transport.	Uncertainty and Modeling Restriction.	Uncertainty distribution for the K_c parameter used in TSPA-SR is for Am on waste form colloids (conservative relative to plutonium) and filtration is not included.
Only Fracture Flow in Volcanic Units (no matrix flow).	Modeling Restriction.	This issue is related to the continuum assumption for fractured units employed in the SZ site-scale flow and transport model. The dual-porosity formulation of the transport model assumes (conservatively) groundwater flow occurs only in the fracture network.

Table F-13. Biosphere

Description of Assumption	Basis for Assumption	Comments
Annual Groundwater Usage (per Draft NRC regulation).	Uncertainty.	Site recommendation water usage based on individual withdrawal permits and not on existing agricultural units in Amargosa Valley. The annual water usage is estimated to be conservative (i.e., lower) by about a factor of two for the farming community proposed by NRC.
Dose Conversion Factors (DCFs).	Uncertainty.	Because of uncertainty in the final chemical/physical form of the radionuclides supplied to the biosphere from groundwater and the uncertainty in the subsequent evolution of the chemical/physical form soil, the pessimistic DCF values for possible gastrointestinal absorption fractions and lung clearance classes for chemical compounds were used.
Transfer Coefficients.	Lack of site specific data.	Value used for the transfer coefficients in the GENII-S code were the conservative values from the published scientific literature. The values used are judged to be either reasonable or conservative for the Amargosa Valley area. (i.e., they are very unlikely to be found non-conservative at a later date.)
Receptor Characteristics.	Lack of receptor specific data.	Behavioral parameters that were not quantified in the survey (such as time per day spent outdoors, and inadvertent soil ingestion, were set to conservative values or ranges of values.
Other GENII-S input parameters.	Uncertainty.	Other GENII-S input parameters that were not quantified in the survey (such as translocation factors, animal feed consumption, fraction of animal feed and water that is contaminated were set to conservative values or ranges of values.
The GENII-S code.	GENII-S sub-model and architecture.	In building the GENII-S code, each sub-model was constructed to err on the side of conservatism to avoid providing under estimates of dose. For example, the amount of radioactivity introduced into the biosphere with irrigation water is accounted for twice; once for deposition on the plant's surfaces and once for root uptake.

Source: GENII-S (CRWMS M&O 1998 [107723])

Table F-14. Disruptive Events

Description of Assumption	Basis for Assumption	Comments
Biosphere dose conversion factor (BDCFs) used for all times after eruptions based on dusty conditions corresponding only to the first few years after the eruption.	Simplification to avoid time-dependent BDCFs in GoldSim.	This may be a significant conservatism.
All eruptions are violent strombolian for their entire duration.	Conservative simplification that is consistent with ASHPLUME capabilities. Data are not readily available to defend a less conservative position.	Eruptions should reasonably be assumed to all have a violent phase for some portion of the eruption. Assuming the entire eruption is violent overestimates the total violent phase release.
All WPs and drip shields in the direct path of the eruption are fully destroyed and all waste is available for transport in the eruption.	Data are not available to defend a less conservative position.	Obtaining data to support package performance in an eruptive environment may not be possible..
Eruptive conduits that intersect drifts in any portion of their diameter are assumed to be centered on the drifts.	Simplification.	This assumption may be reasonable, in that it assumes that eruptive conduits that intersect drifts will spread to involve a segment of the drift as long as the full diameter of the conduit.
All waste (SNF and HLW) in packages in the direct path of the eruption is reduced to small (10s of microns) grain sizes.	Data are not available to defend a less conservative position.	The assumption is probably conservative for HLW, but may be realistic for SNF.
WPs and drip shields adjacent to an intrusive event (i.e., 3 packages on either side) are fully destroyed.	Data are not available to defend a less conservative position.	The assumption probably is conservative, but data are not likely to be available to support other interpretations.
Waste in packages adjacent to intrusive events is fully exposed to ground water transport.	Data are not available to defend a less conservative position.	The assumption probably is conservative, but data are not likely to be available to support other interpretations.
Waste in packages that are destroyed by eruption is double-counted in the groundwater release.	Modeling simplification.	
Wind direction fixed to the south.	Compensation for lack of model for surface redistribution of ash deposits.	This assumption overestimates the direct dose during and following the eruption, and it may overcompensate for the surface redistribution effect.

Source: GoldSim (Golder Associates 2000 [151202]); ASHPLUME (CRWMS M&O 2000 [151349]).

Following the summary of major assumptions and their bases, Table F-15 outlines the aspects of the TSPA-SR model that are treated as uncertain, variable and/or conservative.

Table F-15. Uncertainty, Variability, and Conservatism in the TSPA-SR Model

Key Attributes of System	Process Model	Uncertainty	Variability	Conservatism	Comments
Water Contacting Waste Package	Climate	✓	✓		• Uncertainty is captured by lower and upper bounds for climate. • Variability is captured through timing of climates.
	Net Infiltration	✓	✓		
	UZ Flow	✓	✓	✓	• 10% of flow through Calico Hills vitric unit (which is beneath about half the potential repository) is through the fractures. Results at Busted Butte indicate that 100% of flow through this unit should reside in the matrix.
	Coupled Effects on UZ Flow	✓			• Thermal effects of far-field UZ flow have been screened out.
	Seepage into Emplacement Drifts	✓	✓	✓	• Seepage threshold possibly underestimated. • Seep flux increased by 55% to account for effects of drift degradation & rock bolts.
	Coupled Effects on Seepage			✓	• Percolation flux taken 5 m above drift crown during thermal period as input to seepage model. • Thermal dry-out neglected, possible better performance of lower lithophysal unit neglected.
Waste Package Lifetime	In-Drift Physical and Chemical Environments		✓		• Laboratory A22 corrosion rates measured under extreme chemical environments.
	In-Drift Moisture Distribution		✓	✓	• Threshold RH based on the deliquescence point of NaNO3 salt, and the same threshold RH for both drip and no-drip conditions.
	Drip Shield Degradation and Performance	✓	✓	✓	• Corrosion initiated at threshold RH.
	Waste Package Degradation and Performance	✓	✓	✓	• Corrosion initiated at threshold RH. • Density and orientation of embedded defects/flaws. • Highly aggressive water chemistry conditions for SCC Crack Initiation and Propagation. • Laboratory A22 corrosion rates measured under extreme chemical environments.

Table F-15. Uncertainty, Variability, and Conservatism in the TSPA-SR Model (Continued)

Key Attributes of System	Process Model	Uncertainty	Variability	Conservatism	Comments
Radionuclide Mobilization and Release from the EBS	Radionuclide Inventory			✓	• C-14 included in inventory is transported entirely in the aqueous phase.
	In-Package Environments		✓	✓	• Thermal dry-out effects during first 5-10,000 years neglected. (No effect on DSNF/defense high-level radioactive waste (DHLW.)
	Cladding Degradation and Performance	✓	✓	✓	• Conservative estimate of perforation from creep rupture is used. • Wet unzipping has uncertainty added that is in addition to the uncertainty in the CSNF degradation rate.
	CSNF Degradation and Performance			✓	• Degradation rates do not consider secondary phase formation.
	DSNF Degradation and Performance			✓	• No cladding credit is assumed for DSNF (with the exception of naval SNF). • A constant degradation rate is used that conservatively bounds degradation of metallic uranium present in N-reactor fuel. • Degradation rates do not consider secondary phase formation.
	DHLW Degradation and Performance			✓	• High degradation rates are assumed. • Degradation rates do not consider secondary phase formation.
	Dissolved Radionuclide Concentrations	✓		✓	• High Np solubility is predicted because Np_2O_5 is conservatively assumed to be the controlling solid phase. • Np or other radioisotopes are not incorporated into secondary phases of uranium are assumed.
	Colloid-Associated Radionuclide Concentrations	✓		✓	• Concentration of iron-(hydr)oxide corrosion-product colloids not linked to corrosion rates. • Stabilities of colloid types (waste-form, groundwater, and corrosion-product colloids) are treated independently. • Assumption that groundwater colloids consist of montmorillonite (smectite) clay minerals.

Table F-15. Uncertainty, Variability, and Conservatism in the TSPA-SR Model (Continued)

Key Attributes of System	Process Model	Uncertainty	Variability	Conservatism	Comments
Radionuclide Mobilization and Release from the EBS (Continued)	In-Package Radionuclide Transport			✓	• Diffusion from altered waste form to inner wall of WP neglected.
	EBS (Invert) Degradation and Performance		✓	✓	• The advective flow for radionuclide transport through the invert is a one-dimensional process and always vertically downward. • The binary diffusion coefficient for all radionuclides is bounded by the self-diffusion coefficient of water. • A zero concentration boundary condition at the invert/UZ interface is enforced during the post-closure simulation period.
Transport Away from the EBS	UZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion; Dilution)	✓	✓	✓	• The treatment of fracture/matrix diffusion assumes a stagnant matrix, which has been shown to be conservative. • Colloid concentrations and K_ds used to calculate K_c are conservative (i.e., high). • Colloid retardation is neglected for all colloids in the UZ. • Physical colloid filtration is neglected for transport in fractures.
	SZ Radionuclide Transport	✓	✓	✓	• Sorption coefficients for alluvium is likely low. • No sorption in fractures. • No removal of colloids by mechanical filtration.
	Wellhead Dilution	✓			
	BDCFs	✓		✓	• Pessimistic DCF values for possible gastrointestinal absorption.
Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	✓	✓		• Spatial and temporal variability in igneous processes considered by Probabalistic Volcanic Hazard Analysis panel.
	Characteristics of Volcanic Eruption	✓		✓	• Eruptive events assumed to be violent for full duration.

Table F-15. Uncertainty, Variability, and Conservatism in the TSPA-SR Model (Continued)

Key Attributes of System	Process Model	Uncertainty	Variability	Conservatism	Comments
Effects of Potentially Disruptive Processes and Events (Continued)	Effects of Volcanic Eruption	✓		✓	• Volcanic eruption assumed to degrade all WPs, drip shields, and cladding that are intersected by eruptive conduit.
	Atmospheric Transport of Volcanic Eruption	✓	✓	✓	• Variability in Wind Speed with altitude and time included in cdf. • Assume wind always blows toward critical group (south). (Conservative approach to compensate for not including surface redistribution processes.)
	Biosphere Dose Conversion for Volcanic Eruption	✓		✓	• High air mass loading assumed to persist permanently following ash fall.
	Probability of Igneous Intrusion	✓	✓		• Spatial and temporal variability in igneous processes considered by Probabalistic Volcanic Hazard Analysis panel.
	Characteristics of Igneous Intrusion		✓	✓	• Variability in location (length and orientation) affects extent of damage. • Multiple dikes possible in a single event, assumed to be at least favorable spacing.
	Effects of Igneous Intrusion	✓	✓	✓	• Variability in location (length and orientation) affects extent of damage. • Igneous intrusion assumed to degrade all WP drip shields and cladding in all drifts that are intersected by dike. • Three WPs on either side of the dike are assumed fully degraded.

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APPENDIX G

DATA TRACKING INFORMATION FOR TOTAL SYSTEM PERFORMANCE ASSESSMENT-SITE RECOMMENDATION ANALYSES

APPENDIX G

ACRONYMS AND ABBREVIATIONS

BDCF	biosphere dose conversion factor
CDF	cumulative distribution function
CSNF	commercial spent nuclear fuel
DS	drip shield
DTN	data tracking number
EBS	engineered barrier system
FEHM	finite element heat and mass
GVP	Gaussian variance partitioning
GWPC	ground water protection case
Kd	batch distribution coefficient
MIC	microbiology influenced corrosion
NRC	U.S. Nuclear Regulatory Commission
SZ	saturated zone
TSPA	total system performance assessment
TSPA-SR	Total System Performance Assessment-Site Recommendation
UZ	unsaturated zone
WP	waste package

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APPENDIX G

DATA TRACKING INFORMATION FOR TOTAL SYSTEM PERFORMANCE ASSESSMENT-SITE RECOMMENDATION ANALYSES

Table G-1 lists the simulations that have been conducted for the Total System Performance Assessment-Site Recommendation (TSPA-SR). For each model case, the table provides the run number and run name by which the case is referred to, the basic scenario used, and a description of the run. The table also lists the figures illustrating the model case results, which can be found in this report, and finally the Data Tracking Number (DTN) through which the model data can be accessed in the Technical Data Management System's Model Warehouse. The documentation of the base case model is found in *Total System Performance Assessment (TSPA) Model for Site Recommendation* (CRWMS M&O 2000 [148384]).

Simulations referred to as "base case" involve one of the three scenarios: nominal, igneous (i.e., including volcano event), or human intrusion. The base case description includes the duration of the simulation, and the number of realizations.

Additional analyses modifying the "base case" were also conducted. These are identified as "sensitivity study" in the table. Sensitivity studies were carried by modifying key parameters or models and analyzing the Total System Performance Assessment (TSPA) Model response. The alterations to the base case for the particular sensitivity case are described in the table.

Table G-1. Listing of Simulations Conducted for the TSPA-SR

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
Nominal Cases (Rev 00B)					
Nominal Base Cases					
SR00_047nm5	Base Case 100 Realization	Nominal	Rev 00B Base Case 100 Realizations 1e5 years.	Fig. 3.5-8 Fig. 3.5-12 Fig. 3.5-15 Fig. 3.5-17a Fig. 3.5-17b Fig. 3.5-17c Fig. 3.5-17d Fig. 3.5-20a Fig. 3.5-20b Fig. 4.1-22 Fig 4.6-4 Fig 4.6-5 Fig. 4.6-7 Fig. 4.6-8 Fig. 4.6-9 Fig 4.6-10 Fig. 5.2-1 Fig. 5.2-2a Fig. 5.2-2b Fig. 5.2-3 Fig. 5.2-4 Fig. 5.2-5 Fig. 5.2-6 Fig. 5.2-7 Fig. 5.2-8 Fig. 5.2-9 Fig. 5.2-10	MO0008MWDNM501.005 [151714]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
				Fig. 5.2-11 Fig 5.2-12 Fig. 5.2-13 Fig. 5.2-14 Fig. 5.2-15 Fig. 5.2-16 Fig. 5.3-1 Fig. 5.3-2 Fig. 5.3-3 Fig. 5.3-4 Fig. 5.3-5 Fig. 5.3-6 Fig. 5.3-7 Fig. 5.3-8 Fig. 5.3-9 Fig. 5.3-10 Fig. 5.3-11 Fig. 5.3-12 Fig. 5.3-13	
SR00_023nm6	Base Case 100 Realization	Nominal	Rev 00B Base Case 100 Realizations 1e6 years.	N/A	MO0009MWDNM601.018 [152839]
SR00_049nm5	Base Case 300 Realization	Nominal	Rev 00B Base Case 300 Realizations 1e5 years.	Fig. 4.1-5 Fig. 4.1-8 Fig. 4.1-9 Fig. 4.1-14 Fig. 4.1-16 Fig. 4.1-17 Fig. 4.1-18 Fig. 4.1-22 Fig. 5.1-1	MO0007MWDTSP01.002 [151716]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
				Fig. 6.1-1	
			Post GoldSim Uncertainty Importance Runs.	Fig. 5.1-2 Fig. 5.1-3 Fig. 5.1-4 Fig. 5.1-5 Fig. 5.1-6 Fig. 5.1-7 Fig. 5.1-8 Fig. 5.1-9 Fig. 5.1-10 Fig. 5.1-12 Fig. 5.1-13 Fig. 5.1-14 Fig. 5.1-15 Fig. 5.1-16 Fig. 5.1-17	MO0009MWDTSP01.019 [153131]
SR00_091nm5	Base Case 300 Realization Rev00B1	Nominal	Rev 00B1 Base Case 300 Realizations 1e5 years. Same as SR00-049 nm5 except additional data has been saved	Fig 4.1-6 Fig 4.1-7 Fig 4.1-11 Fig 4.1-12 Fig 4.1-13 Fig 4.1-15 Fig 4.3-3 Fig. 6.1-3	MO0007MWDTSP01.003 [151706]
SR00_108nm5	500 Realization Base Case 1e5	Nominal	500 Realization Base Case 1e5, derived from Rev. 00B1.	Fig. 4.1-22	MO0009MWDNM501.017 [153132]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_161nm5	Engineered Barrier System (EBS) Only Run for Chemistry Plots	Nominal	Run Base Case with EBS only and saving all chemistry for analysis plots, 300 Realizations.	Fig. 3.5-6 Fig. 3.5-14 Fig. 4.1-10	MO0008MWDNM501.007 [151719]
Ground Water Protection Case					
SR00_042nm6	Ground Water Protection Case (GWPC) 300 Realization 1e6 with Fixed Ingrowth and New Saturated Zone (SZ) Curves	Nominal	300 realizations 1e6, additional radionuclides transported through finite element heat and mass (FEHM) (^{230}Th and ^{232}Th); additional nuclides included in dose (^{230}Th, ^{226}Ra, ^{210}Ph, ^{232}Th, ^{228}Ra); total Ra conc. in groundwater; gross alpha activity in groundwater; additional biosphere dose conversion factors (BDCFs) required; Calculate Nominal dose with additional radionuclides (^{230}Th, ^{232}Th, ^{226}Ra, ^{228}Ra, ^{210}Pb). ^{232}Th and ^{234}U were not correctly decayed to daughter products in the EBS for case SR00_149nm5, SR00_010nm6, SR00_015nm5, and SR00_019nm6. ^{232}Th decays to ^{228}Ra and ^{234}U decays to ^{230}Th. For the unsaturated zone (UZ) input file ptrk.multi, ^{234}U was corrected to decay to ^{230}Th as well. Additionally, for the calculation of dose for Th species, the SZ one-dimensional model was used instead of the SZ three-dimensional model. Set PTRK file to decay w/ingrowth for ^{230}Th, rather then just simple decay of ^{234}U. Used new SZ curves to eliminate mass generation in the SZ for select RNs over several realizations. Re-run SR00_021nm6 with fixed Pu242 BDCF.	Fig. 4.1-19a Fig. 4.1-19b Fig. 4.1-20 Fig. 4.1-21 Fig. 4.1-23 Fig. 4.1-24 Fig. 4.1-25 Fig. 4.1-26 Fig. 4.1-27 Fig. 6.1-6 Fig. 6.1-7 Fig. 6.1-8	MO0011MWDNM601.021 [153126]
			Post GoldSim Uncertainty Importance Runs.	Fig. 5.1-11	MO0011MWDREG01.001
SR00_035nm6	Secondary solubilities with fixed Pu242 BDCF	Nominal	Re-run of SR00_027nm6 with corrected Pu242 BDCF.	Fig. 4.1-20 Fig. 4.1-21 Fig. 5.2-12	MO0011MWDMIL01.022 [153128]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_034nm6	Secondary solubilities and long-term climate, with fixed Pu242 BDCF.	Nominal	Re-run of SR00_031nm6 with corrected Pu242 BDCF.	Fig. 4.1-21	MO0011MWDMIL01.022 [153128]
Barrier Sensitivity Cases					
SR00_072nm5	300 Realization WAPDEG Sens	Nominal	Set z.OL, z.ML,U.GVP.GA22x2P5.CDF, U.GVP.GA22SR00; qu.GVP.GA22x2P5.CDF and qu.GVP.GA22SR00 at Median Values. – Then perform Post GoldSim Uncertainty Importance Runs.	Figure 5.1-18	MO0008MWDBARRI.000 [152184]
SR00_050nm5	Waste Package (WP) Degraded Barrier	Nominal	95th percentile WAPDEG WP Failure Degraded Barrier case; 100 Realizations: 1) G(MIC), microbiology influenced corrosion (MIC) enhancement factor - 1.95 2) G(age), aging enhancement factor - 2.425 3) Set U's in (Gaussian variance partitioning (GVP) 1, 3, and 5 calls in \WP_Degradation\GVP_External\ to 0.05 and all qu's to 0.95 4) stress/stress intensity profile (95th percentile from +/- 30% uncertainty range); z_OL and z_ML set to = 2.05132 5) manufacturing defect probability (or defect number per WP) - b = 4.83, v = 2.9, Psi = 0.362445	Figure 5.3-5 Figure 5.3-6	MO0008MWDBARRI.000 [152184]
SR00_064nm5	WP Enhanced Barrier	Nominal	5th percentile WAPDEG WP Failure Improved Barrier Case, 100 Realizations: 1) G(MIC), MIC enhancement factor = 1.05 2) G(age), aging enhancement factor = 1.075 3) Set U's in GVP 1, 3, and 5 calls in \WP_Degradation\GVP_External\ to 0.95 and all qu's to 0.05 4) stress/stress intensity profile (5th percentile from +/- 30% uncertainty range); z_OL and z_ML set to = -2.05132	Figure 5.3-5 Figure 5.3-6	MO0008MWDBARRI.000 [152184]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
			5) manufacturing defect probability (or defect number per WP): b = 1.77, v = 1.1, Psi = 0.348855		
SR00_061nm5	Drip Shield (DS) Degraded Barrier	Nominal	95th percentile WAPDEG DS Failure Case, 100 Realizations: The GVP 2 and GVP 4 parameters U were fixed at 0.05. The GVP 2 and GVP 4 parameters qu were fixed at 0.95.	Figure 5.3-3 Figure 5.3-4	MO0008MWDBARRI.000 [152184]
SR00_065nm5	DS Enhanced Barrier	Nominal	5th percentile WAPDEG DS Failure Case, 100 Realizations: Set the GVP 2 and GVP 4 U parameters to 0.95. Set the GVP 2 and GVP 4 qu parameters to 0.05.	Figure 5.3-3 Figure 5.3-4	MO0008MWDBARRI.000 [152184]
SR00_066nm5	Cladding Degraded Barrier	Nominal	95th percentile Cladding Values on clad stochastics (4 parameters; unzip_uncertainty, uncert_a0; LC_uncert; Initial_Rod_Failures).	Figure 5.3-7	MO0008MWDBARRI.000 [152184]
SR00_067nm5	Cladding Enhanced Barrier	Nominal	5th percentile Cladding Values on clad stochastics (4 parameters; uzip_uncertainty, uncert_a0; LC_uncert; Initial_Rod_Failures).	Figure 5.3-7	MO0008MWDBARRI.000 [152184]
SR00_062nm5	UZ Flow Degraded Barrier (entire system)	Nominal	High infiltration throughout model (set infiltration_scenario = 3).	Figure 5.2.1	MO0008MWDBARRI.000 [152184]
SR00_063nm5	UZ Flow Enhanced Barrier (entire system)	Nominal	Low infiltration throughout model (set infiltration_scenario = 1).	Figure 5.2.1	MO0008MWDBARRI.000 [152184]
SR00_126nm5	UZ Flow and Transport Degraded Barrier (entire system)	Nominal	95th percentile UZ transport parameters (UZ_Params_Multi.sr File), High Infiltration UZ flow for the entire system (set infiltration_scenario = 3).	Figure 5.3-11	MO0008MWDBARRI.000 [152184]
SR00_125nm5	UZ Flow and Transport Enhanced Barrier (entire system)	Nominal	5th percentile UZ transport parameters (UZ_Params_Multi.sr File); Low Infiltration UZ Flow for the entire system (set infiltration_scenario = 1).	Figure 5.3-11	MO0008MWDBARRI.000 [152184]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_087nm5	SZ Flow and Transport Degraded Barrier	Nominal	FPLAN=0.05, NVF19=0.0962, FISVO=1.99, FPVO=-4.8, DCVO=-12.85, KDNPVO=0.0031, KDNPAL=0.202, KDIAL=0.335, KDUVO=0.2, KDUAL=0.4, GWSPD=0.95, KDRN10=0.95, KDRN9=0.95, CORAL=0.0403, CORVO=0.233, HAVO=0.25, LDISP=2., KDTCAL=0.287, KC_AM_GW_COLLOID=0.129; Used SZ breakthrough curves for 5th/95th percentile runs, set SZ_Index = 1 for 95th percentile run.	Figure 5.3-13	MO0008MWDBARRI.000 [152184]
SR00_084nm5	SZ Flow and Transport Enhanced Barrier	Nominal	FPLAN=0.95, NVF19=0.2639, FISVO=0.588, FPVO=-1.2, DCVO=-10.15, KDNPVO=1.542, KDNPAL=57.99, KDIAL=0.6145, KDUVO=3.8, KDUAL=7.6, GWSPD=0.05, KDRN10=0.05, KDRN9=0.05, CORAL=3.8743, CORVO=2.8079, HAVO=0.75, LDISP=2., KDTCAL=0.6025, KC_AM_GW_COLLOID=5.72e-5; Used SZ breakthrough curves for 5th/95th percentile runs, set SZ_Index = 2 for 5th percentile run.	Figure 5.3-13	MO0008MWDBARRI.000 [152184]
SR00_070nm5	EBS Transport Degraded Barrier	Nominal	Use Rev00a diffusion model (ϕ^1, S^1, no uncertainty term) and 95th percentile Radionuclide concentration case (95th solubilities, WP Chem in invert, max colloid stability, 95th batch distribution coefficient (Kd colloids).	Figure 5.3-9	MO0008MWDBARRI.000 [152184]
SR00_145nm5	EBS Transport Enhanced Barrier	Nominal	Use 1e-11 cm2/sec diffusion model and 5th percentile Radionuclide concentration case (5th solubilities, Invert Chem in invert, min colloid stability, 5th Kd colloids).	Figure 5.3-9	MO0008MWDBARRI.000 [152184]
SR00_130nm5	Seepage Degraded Barrier	Nominal	Set flow focus factor to 95th (9.5), set seepage uncert. to 95th (0.95).	Figure 5.3-1	MO0008MWDBARRI.000 [152184]
SR00_129nm5	Seepage Enhanced Barrier	Nominal	Set flow focus factor to 5th (1.21267), set seepage uncert. to 5th (0.05).	Figure 5.3-1	MO0008MWDBARRI.000 [152184]
SR00_128nm5	Seepage degraded + UZ Flow Degraded + UZ Transport Degraded	Nominal	Seepage degraded, high infiltration UZ flow throughout entire model set (infiltration_scenario = 3), 95th percentile UZ transport parameters (UZ_Params_Multi.sr file), set flow focus factor to 95th (9.5), set seepage uncert. to 95th (0.95).	Figure 5.3-12	MO0008MWDBARRI.000 [152184]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_127nm5	Seepage enhanced + UZ Flow enhanced + UZ Transport enhanced	Nominal	Seepage enhanced, low infiltration UZ flow throughout entire model set (infiltration_scenario = 1), 5^{th} percentile UZ transport parameters (UZ_Params_Multi.sr file), set flow focus factor to 5^{th} (1.21267), set seepage uncert. to 5^{th} (0.05).	Figure 5.3-12	MO0008MWDBARRI.000 [152184]
SR00_076nm5	Seepage degraded + UZ Flow Degraded	Nominal	Seepage degraded, high infiltration UZ flow throughout entire model, (infiltration scenario = 3), set flow focus factor to 95^{th} (9.5), set seepage uncert. to 95^{th} (0.95).	Figure 5.3-2	MO0008MWDBARRI.000 [152184]
SR00_075nm5	Seepage enhanced + UZ Flow enhanced	Nominal	Seepage enhanced, low infiltration UZ flow throughout entire model, (infiltration scenario = 1), set flow focus factor to 5^{th} (9.5), set seepage uncert. to 5^{th} (0.05).	Figure 5.3-2	MO0008MWDBARRI.000 [152184]
SR00_143nm5	Biosphere 95^{th}	Nominal	95^{th} percentile BDCFs Values.	Figure 5.2-15	MO0008MWDBARRI.000 [152184]
SR00_144nm5	Biosphere 5^{th}	Nominal	5^{th} percentile BDCFs Values.	Figure 5.2-15	MO0008MWDBARRI.000 [152184]
SR00_048nm5	Backfill Sensitivity Case	Nominal	Base Case with Backfill. 100 realizations 1e5.	Fig 4.6-4 Fig 4.6-5	MO0008MWDBARRI.000 [152184]
SR00_094nm5	Concentration Limits Degraded Barrier	Nominal	95^{th} percentile Solubilities, WP Chem in the invert, max colloid stability, 95^{th} percentile Kd colloids.	Figure 5.3-8	MO0008MWDBARRI.000 [152184]
SR00_093nm5	Concentration Limits Enhanced Barrier	Nominal	5^{th} percentile Solubilities, Invert Chem in the invert, min colloid stability, 5^{th} percentile Kd colloids.	Figure 5.3-8	MO0008MWDBARRI.000 [152184]
SR00_095nm5	Rev. 00A Diffusion Model	Nominal	REV 00A diffusion model (ϕ^1, S^1, no uncertainty term).	Figure 5.2-13	MO0008MWDBARRI.000 [152184]
SR00_131nm5	Low Diffusion Model	Nominal	1e-11 cm^2/sec diffusion model.	Figure 5.2-13	MO0008MWDBARRI.000 [152184]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_122nm5	UZ Transport Degraded Barrier	Nominal	95^{th} percentile UZ Transport Parameters (UZ_Params_Multi.sr file changes): - 5^{th} percentile (i.e., low) Kds for all nuclides. - 95^{th} percentile Kc. - 5^{th} percentile anion/cation matrix diffusion coefficients. - 95^{th} percentile fracture apertures for all.	Figure 5.3-10	MO0008MWDBARRI.000 [152184]
SR00_123nm5	UZ Transport Enhanced Barrier	Nominal	5^{th} percentile UZ Transport Parameters (UZ_Params_Multi.sr file changes): - 95^{th} percentile (i.e., low) Kds for all nuclides. - 5^{th} percentile Kc. - 95^{th} percentile anion/cation matrix diffusion coefficients. - 5^{th} percentile fracture apertures for all.	Figure 5.3-10	MO0008MWDBARRI.000 [152184]
SR00_056nm5	WAPDEG Sens #1	Nominal	10-40% Stress Threshold, Yield Strength.	Figure 5.2-5 Figure 5.2-6	MO0008MWDBARRI.000 [152184]
SR00_057nm5	WAPDEG Sens #3	Nominal	10-40% Stress Threshold, Stress Uncertainty =±10% Yield Strength (0.3-1.0 fys).	Figure 5.2-5 Figure 5.2-6	MO0008MWDBARRI.000 [152184]
SR00_052nm5	WAPDEG Sens #4	Nominal	Stress Uncertainty =±10% Yield Strength (0.3-1.0 fys).	Figure 5.2-5 Figure 5.2-6	MO0008MWDBARRI.000 [152184]
SR00_053nm5	WAPDEG Sens #5	Nominal	10-40% Stress Threshold; Stress Uncertainty =±10% Yield Strength (0.3-1.0 fys); and Radial Defects 1% Total Defects (psi * 0.1.).	Figure 5.2-5 Figure 5.2-6	MO0008MWDBARRI.000 [152184]
SR00_116nm5	Co-disposal Waste Package Sensitivity	Nominal	Run nominal scenario with only co-disposal packages failing.	Figure 5.2-11	MO0008MWDBARRI.000 [152184]
SR00_109nm5	Commercial Spent Nuclear Fuel (CSNF) WPs Sensitivity	Nominal	Run nominal scenario with only co-disposal packages failing.	Figure 5.2-11	MO0008MWDBARRI.000 [152184]
SR00_106nm5	Seepage Flow Focus 95^{th} percentile	Nominal	Set seepage flow focus factor to 95^{th} percentile value (9.5).	Figure 5.2-2a	MO0008MWDBARRI.000 [152184]
SR00_107nm5	Seepage Flow Focus 5^{th} percentile	Nominal	Set seepage flow focus factor to 5^{th} percentile value (1.21267).	Figure 5.2-2a	MO0008MWDBARRI.000 [152184]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_147nm5	Water Usage (95^{th})	Nominal	Set Annual_Ground_Water_Usage to 95^{th} percentile value (3.49504×10^6 m^3/yr).	Figure 5.2-16	MO0008MWDBARRI.000 [152184]
SR00_148nm5	Water Usage (5^{th})	Nominal	Set Annual_Ground_Water_Usage to 5^{th} percentile value (1.47989×10^6 m^3/yr).	Figure 5.2-16	MO0008MWDBARRI.000 [152184]
SR00_159nm5	Seepage Degraded Barrier (no flow focus, seepage fraction = 1.0, and seepage uncert = 0.95)	Nominal	No flow focus factor, local seepage fraction set to 1.0, seepage uncertainty set to 0.95.	Figure 5.2-2b	MO0008MWDBARRI.000 [152184]
SR00_160nm5	Seepage Degraded Barrier (no flow focus, seepage fraction = 1.0)	Nominal	No flow focus factor, local seepage fraction set to 1.0.	Figure 5.2-2b	MO0008MWDBARRI.000 [152184]
SR00_165nm5	UZ Matrix Diffusion Sensitivity (100 Times)	Nominal	UZ Matrix Diffusion. UZ anions and cations were set to 100 times sampled values for 100 realizations.	Figure 5.2-14	MO0008MWDBARRI.000 [152184]
SR00_166nm5	UZ Matrix Diffusion Sensitivity (0 Times)	Nominal	UZ Matrix Diffusion. UZ anions and cations were set to 0 times sampled values for 100 Realizations.	Figure 5.2-14	MO0008MWDBARRI.000 [152184]
SR00_077nm5	z.OL and z.ML 95^{th}	Nominal	set z.OL and z.ML to 95^{th} percentile values.	Figure 5.2-3 Figure 5.2-4	MO0008MWDBARRI.000 [152184]
SR00_078nm5	z.OL and z.ML 5^{th}	Nominal	set z.OL and z.ML to 5^{th} percentile values.	Figure 5.2-3 Figure 5.2-4	MO0008MWDBARRI.000 [152184]
SR00_079nm5	qu 's at 95^{th}	Nominal	set qu.GVP.GA22x2P5.CDF and qu.GVP.GA22SR00 to 95^{th} percentile values.	Figure 5.2-9 Figure 5.2-10	MO0008MWDBARRI.000 [152184]
SR00_080nm5	qu 's at 5^{th}	Nominal	set qu.GVP.GA22x2P5.CDF and qu.GVP.GA22SR00 to 5^{th} percentile values.	Figure 5.2-9 Figure 5.2-10	MO0008MWDBARRI.000 [152184]
SR00_082nm5	U's at 95^{th}	Nominal	set U.GVP.GA22x2P5.CDF and U.GVP.GA22SR00 to 95^{th} percentile values.	Figure 5.2-7 Figure 5.2-8	MO0008MWDBARRI.000 [152184]
SR00_081nm5	U's at 5^{th}	Nominal	set U.GVP.GA22x2P5.CDF and U.GVP.GA22SR00 to 5^{th} percentile values.	Figure 5.2-7 Figure 5.2-8	MO0008MWDBARRI.000 [152184]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
Igneous Cases (REV 00B)					
SR00_016im5	Base Case 300 Realization	Igneous	300 Realization Base Case Run.	Fig. 3.10-14 Fig. 3.10-15 Figure 5.2-18 Figure 5.2-19 Figure 5.2-20 Figure 5.2-21 Figure 5.2-22 Figure 5.2-25 Figure 5.2-26	MO0008MWDIM501.006 [151720]
SR00_044im5	Igneous 1,000 Realization Base Case	Igneous	1,000 Realization Base Case for 100,000 years.	Figure 4.2-5	MO0008MWDIGNEO.000 [152185]
SR00_002im4	Igneous 1,000 Realization Base Case	Igneous	1,000 Realization Base Case for 20,000 years.	Figure 4.2-5 Figure 4.6-6 Figure 5.2-17 Figure 5.2-23 Figure 5.2-24	MO0008MWDIGNEO.000 [152185]
SR00_005im4	Igneous 5,000 Realization Base Case	Igneous	5,000 Realization Base Case for 50,000 years.	Figure 4.2-1 Figure 4.2-2 Figure 4.2-3 Figure 4.2-4 Figure 4.2-5 Figure 4.3-3 Figure 6.1-2 Figure 6.1-3	MO0009MWDIM401.015 [153123]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
			Post GoldSim Uncertainty Importance Runs	Figure 5.1-19 Figure 5.1-20 Figure 5.1-21 Figure 5.1-22	MO0011MWDRE60V.001
SR00_039im5	Base Case 300 Realizations w/Backfill	Igneous	300 Realizations Base Case Run with backfill.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_022im5	Combined Zone 2 Aperture 95th and Zone 1, Zone 2 packages hit CDF 95th	Igneous	95th percentile Zone 2 package apertures, and 95th percentile Zone 1 and Zone 2 packages hit.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_031im5	Conduit Diameter 95th	Igneous	95th percentile Conduit Diameter.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_033im5	Mean Ash Particle Diameter 95th	Igneous	95th percentile Mean Ash Particle Diameter.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_024im5	1,000 year Soil Removal Factor	Igneous	1,000 year Soil Removal Factor.	Figure 5.2-20	MO0008MWDIGNEO.000 [152185]
SR00_019im5	U.S. Nuclear Regulatory Commission (NRC) 1e-7 Prob.	Igneous	NRC 1e-7 Event Probability Run.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_017im5	Zone 1 & Zone 2 Cumulative Distribution Function (CDF) 95th	Igneous	Sample 95th percentile CDFs for number of damaged WPs (Zone 1 and Zone 2).	N/A	MO0008MWDIGNEO.000 [152185]
SR00_018im5	Zone 1 & Zone 2 CDF 5th	Igneous	Sample 5th percentile CDFs for number of damaged WPs (Zone 1 and Zone 2).	N/A	MO0008MWDIGNEO.000 [152185]
SR00_030im5	Eruptive Volume 95th	Igneous	Sample 95th percentile Value for Eruptive Volume.	Figure 5.2-21	MO0008MWDIGNEO.000 [152185]
SR00_027im5	Eruptive Volume 5th	Igneous	Sample 5th percentile Value for Eruptive Volume.	Figure 5.2-21	MO0008MWDIGNEO.000 [152185]
SR00_035im5	No Soil Removal Factor	Igneous	No soil removal factor.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_028im5	Wind Speed 95th	Igneous	Sample Wind Speed CDF at 95th percentile value.	Figure 5.2-19	MO0008MWDIGNEO.000 [152185]
SR00_029im5	Wind Speed 5th	Igneous	Sample Wind Speed CDF at 5th percentile value.	Figure 5.2-19	MO0008MWDIGNEO.000 [152185]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_034im5	Mean Ash Particle 5^{th}	Igneous	Sample Mean Ash Particle Diameter at 5^{th} percentile value.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_026im5	Igneous BDCF's 95^{th}	Igneous	Sample Igneous BDCF's at the 95^{th} percentile value.	Figure 5.2-25	MO0008MWDIGNEO.000 [152185]
SR00_025im5	Igneous BDCF's 5^{th}	Igneous	Sample Igneous BDCF's at the 5^{th} percentile value.	Figure 5.2-25	MO0008MWDIGNEO.000 [152185]
SR00_032im5	Conduit Diameter 5^{th}	Igneous	Sample Conduit diameter at the 5^{th} percentile value.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_023im5	Combined 5^{th} Zone 2 Aperture and 5^{th} Zone 1 and Zone 2 Packages Damaged CDF	Igneous	Sample Zone 2 apertures at 5^{th} percentile value, sample both Zone 1 and Zone 2 number of waste packages damaged CDF at the 5^{th}percentile value.	N/A	MO0008MWDIGNEO.000 [152185]
SR00_036im5	95^{th}percentile Number of Direct Release Packages Hit	Igneous	Sample Direct Release Packages Hit (using Base Case Igneous Results) at the 95^{th} percentile value.	Figure 5.2-22	MO0008MWDIGNEO.000 [152185]
SR00_037im5	5^{th}percentile Number of Direct Release Packages Hit	Igneous	Sample Direct Release Packages Hit (using Base Case Igneous Results) at the 5^{th} percentile value.	Figure 5.2-22	MO0008MWDIGNEO.000 [152185]
SR00_038im5	Variable Wind Direction	Igneous	Used sampled wind direction value.	Figure 3.10-14 Figure 3.10-15 Figure 5.2-18	MO0008MWDIGNEO.000 [152185]
SROO_043iM5	Incorporation Ratio set 1	Igneous	Run 300 Realization Igneous scenario with incorporation ratio = .1 (Rhocut).	Figure 5.2-26	MO0008MWDIGNEO.000 [152185]
SR00_047im5	Incorporation Ratio set to 0.1	Igneous	Run 300 Realization Igneous scenario with incorporation ratio = .1 (Rhocut).	Figure 5.2-26	MO0008MWDIGNEO.000 [152185]
SR00_009im4	Igneous Base Case w/backfill 1,000 rlz/20k	Igneous	1,000 Realization igneous base case w/backfill run for 20,000 years.	4.6-6	MO0008MWDIGNEO.000 [152185]
SR00_003im4	95^{th}percentile Number of Zone 2 Packages Hit 1,000 rlz/20k	Igneous	1,000 Realization igneous run for 20,000 years. Set the CDF for Zone 2 + Zone1 packages hit to 95^{th} percentile value.	Figure 5.2-23	MO0008MWDIGNEO.000 [152185]
SR00_006im4	5^{th}percentile Number of Zone 2 Packages Hit 1,000 rlz/20k	Igneous	1,000 Realization igneous run for 20,000 years. Set the CDF for Zone 2 + Zone1 packages hit to 5^{th} percentile value.	Figure 5.2-23	MO0008MWDIGNEO.000 [152185]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_007im4	Combined 95thpercentile Number of Zone 2 Packages Hit and Zone 2 Apertures 1,000 rlz/20k	Igneous	1,000 Realization igneous run for 20,000 years. Set the CDF for Zone 2 + Zone1 packages hit to 95th percentile value. Set zone 2 aperture sizes to the 95th percentile values.	Figure 5.2-24	MO0008MWDIGNEO.000 [152185]
SR00_008im4	Combined 5thpercentile Number of Zone 2 Packages Hit and Zone 2 Apertures 1,000 rlz/20k	Igneous	1,000 Realization igneous run for 20,000 years. Set the CDF for Zone 2 + Zone1 packages hit to 5th percentile value. Set zone 2 aperture sizes to the 5th percentile values.	Figure 5.2-24	MO0008MWDIGNEO.000 [152185]
SR00_004im4	Combined NRC event probability 1e-7/year, vent probability = 1. 1,000 realizations/20k	Igneous	1,000 Realization igneous run for 20,000 years. Set the event probability to 1e-7 per year and set the vent probability factor equal to 1.	Figure 5.2-17	MO0008MWDIGNEO.000 [152185]
Human Intrusion					
SR00_005hm5	Human Intrusion 1e5, 100 realizations, CSNF WP failed at 100 years	H.I.	100 Realizations, derived from Rev. 00B of nominal base case,WP failed at 100 years; transport additional radionuclides through SZ (^{90}Sr, ^{137}Cs, ^{238}Pu, ^{241}Am).	Figure 4.4-12 Figure 5.2-27 Figure 6.1-5	MO0008MWDHUMAN.000 [152186]
SR00_006hm5	Human Intrusion 1e5, 300 realizations, CSNF WP failed at 100 years	H.I.	300 Realizations, derived from Rev. 00B of nominal, WP failed at 100 years; transport additional radionuclides through SZ (^{90}Sr, ^{137}Cs, ^{238}Pu, ^{241}Am).	Figure 4.4-11 Figure 6.1-4	MO0008MWDHUMAN.000 [152186]
SR00_007hm5	Human Intrusion 1e4 Failure	H.I.	Human Intrusion Scenario with 1e4 year WP breach.	Figure 4.4-12 Figure 6.1-5	MO0008MWDHUMAN.000 [152186]
SR00_008hm5	Human Intrusion 95thpercentile Seepage	H.I.	Human Intrusion Scenario with 95th percentile borehole seepage value.	Figure 5.2-27	MO0008MWDHUMAN.000 [152186]
RSS4 Neutralization Runs					
SR00_099nm5	Neutralize DS	Nominal	Remove DS, set DS_Frac_Patch = 1.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_089nm5	Neutralize SZ	Nominal	Feed UZ releases directly to biosphere cell.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_090nm5	Neutralize SZ + UZ	Nominal	Feed EBS releases directly to biosphere cell.		MO0008MWDNEUTR.000 [152187]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_121nm5	Neutralize UZ	Nominal	Feed EBS releases directly to SZ.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_097nm5	Neutralize WP	Nominal	WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_096nm5	Neutralize WP and DS	Nominal	WP Neutralization Case, DS Neutralization Case	N/A	MO0008MWDNEUTR.000 [152187]
SR00_100nm5	Neutralize overlying rock	Nominal	Set seepage flux equal to precipitation, as a function of climate and infiltration scenario: Low / Medium / High Present day 191.6 196.9 277.5(mm/yr) Moonsoon 196.9 309.3 421.6(mm/yr) Glacial trans. 205.5 323.1 440.6(mm/yr)	N/A	MO0008MWDNEUTR.000 [152187]
SR00_124nm5	Neutralize Invert	Nominal	100% invert saturation, WP Chemistry in invert.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_088nm5	No Cladding	Nominal	Use only CSNF degradation rate instead of unzipping, fail all pins.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_113nm5	Neutralize WP and Cladding	Nominal	WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally. Use only CSNF degradation rate instead of unzipping, fail all pins.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_112nm5	Neutralize WP, DS, and Cladding	Nominal	WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally. Use only CSNF degradation rate instead of unzipping, fail all pins. Remove DS.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_114nm5	Neutralize UZ, SZ, and Overlying Rock	Nominal	Feed EBS releases directly to biosphere cell. Set seepage flux equal to precipitation, as a function of climate and infiltration scenario: Low / Medium / High Present day 191.6 196.9 277.5 (mm/yr) Monsoon 196.9 309.3 421.6 (mm/yr) Glacial trans. 295.5 323.1 440.6 (mm/yr)	N/A	MO0008MWDNEUTR.000 [152187]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_119nm5	Neutralize UZ, SZ, Overlying Rock, WPs, DS, Invert, and Cladding	Nominal	WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally. Use only CSNF degradation rate instead of unzipping, fail all pins. Remove DS. Feed EBS releases directly to biosphere cell, Set seepage flux equal to precipitation, as a function of climate and infiltration scenario: Low / Medium / High Present day: 191.6 / 196.9 / 277.5 (mm/yr) Monsoon: 196.9 / 309.3 / 421.6 (mm/yr) Glacial trans.: 205.5 / 323.1 / 440.6 (mm/yr)	N/A	MO0008MWDNEUTR.000 [152187]
SR00_111nm5	Neutralize Concentration Limits	Nominal	No solubility limits, radionuclide mobilization controlled only by degradation rates and cladding.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_115nm5	Neutralized Concentration Limits and WP	Nominal	No solubility limits, radionuclide mobilization controlled only by degradation rates and cladding. WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally.	N/A	MO0008MWDNEUTR.000 [152187]
SR00_132nm5	Neutralized WP with Low Diffusion	Nominal	WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally. Use 1e-11 cm2/sec invert diffusion.	N/A	MO0008MWDNEUTR.000 [152187]
Juvenile Failure Runs					
SR00_153nm5	Juvenile Failure (co-disposal)	Nominal	Single co-disposal WP breach with one patch at 100 years.	N/A	MO0008MWDJUVEN.000 [152188]
SR00_117nm5	Juvenile Failure (CSNF)	Nominal	Single CSNF WP breach with one patch at 100 years.	N/A	MO0008MWDJUVEN.000 [152187]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_137nm5	Juvenile Failure and Degraded DS	Nominal	Fail Single WP with one patch at 100 years, Degraded WAPDEG DS Failure Case, 100 Realizations. The GVP 2 and GVP 4 parameters U were fixed at 0.05. The GVP 2 and GVP 4 parameters qU were fixed at 0.95.	N/A	MO0008MWDJUVEN.000 [152188]
SR00_146nm5	Juvenile Failure and Degraded Cladding Case	Nominal	Fail Single WP with one patch at 100 years, 95th percentile Cladding values on clad stochastics (4 parameters; uzip_uncertainty, uncert_a0; LC_uncert; Initial_Rod_Failures).	N/A	MO0008MWDJUVEN.000 [152188]
SR00_138nm5	Juvenile Failure and Degraded Concentration Limits	Nominal	Fail Single WP with one patch at 100 years, 95th percentile Solubilities, WP Chem in the invert, max colloid stability, 95th percentile Kd colloids.	N/A	MO0008MWDJUVEN.000 [152188]
SR00_139nm5	Juvenile Failure and UZ Transport 95th percentile	Nominal	Fail Single WP with one patch at 100 years, 95th percentile UZ transport parameters (UZ_Params_Multi.sr file modified).	N/A	MO0008MWDJUVEN.000 [152188]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_140nm5	Juvenile Failure and Degraded SZ Flow and Transport	Nominal	Fail Single WP with one patch at 100 years, 95th percentile SZ break through curves, FPLAN=0.05, NVF19=0.0962, FISVO=1.99, FPVO=-4.8, DCVO=-12.85, KDNPVO=0.0031, KDNPAL=0.202, KDIAL=0.335, KDUVO=0.2, KDUAL=0.4, GWSPD=0.95, KDRN10=0.95, KDRN9=0.95, CORAL=0.0403, CORVO=0.233, HAVO=0.25, LDISP=2., KDTCAL=0.287, KC_AM_GW_COLLOID=0.129; Used SZ breakthrough curves for 5th/95th percentile runs, set SZ_Index = 2 for 5th percentile run.	N/A	MO0008MWDJUVEN.000 [152188]
SR00_134nm5	Juvenile Failure and Degraded Seepage & Degraded UZ Flow	Nominal	Fail Single WP with one patch at 100 years, seepage degraded, high infiltration UZ flow throughout entire model (set infiltration scenario = 3), set flow focus factor to 95th (9.5), set seepage uncert. to 95th (0.95).	N/A	MO0008MWDJUVEN.000 [152188]
SR00_135nm5	Juvenile Failure and Degraded Seepage & Degraded UZ Flow, Low Diffusion	Nominal	Fail Single WP with one patch at 100 years, seepage degraded, high infiltration UZ flow throughout entire model (set infiltration scenario = 3), set flow focus factor to 95th (9.5), set seepage uncert. to 95th (0.95). Use 1e-11 cm2/sec diffusion.	N/A	MO0008MWDJUVEN.000 [152188]
SR00_136nm5	Juvenile Failure and Enhanced Seepage & Enhanced UZ Flow, Low Diffusion	Nominal	Fail Single WP with one patch at 100 years, seepage enhanced, low infiltration UZ flow throughout entire model (set infiltration scenario = 1), set flow focus factor to 5th (1.21267), set seepage uncert. to 5th (0.05). Use 1e-11 cm2/sec diffusion.	N/A	MO0008MWDJUVEN.000 [152188]
SR00_141nm5	Juvenile Failure CSNF	Nominal	300 Realization CSNF Juvenile Failure Case.	N/A	MO0008MWDJUVEN.000 [152188]

Table G-1. Listing of Simulations Conducted for the TSPA-SR (Continued)

Simulation Number	Simulation Name	Scenario	Description	Figures	DTN
SR00_152nm5	Neutralize all barriers	Nominal	WP Neutralization Case, fail all WPs at T=100 years with one patch, have WAPDEG degradation proceed normally. Use only CSNF degradation rate instead of unzipping, fail all pins. Remove DS. Set seepage flux equal to precipitation, as a function of climate. Compute weighted average concentration in waste-form cells and multiply by BDCFs to get the dose rate. Low / Medium / High Present-day 191.6 196.9 277.5(mm/yr) Monsoon 196.9 309.3 421.6 (mm/yr) Glacial trans. 205.5 323.1 440.6 (mm/yr)	N/A	MO0008MWDJUVEN.000 [152188]
Low Thermal Load					
SR00_180nm5	T/H Sensitivity #4 (0.90/100 Temperature Case)	Nominal	100 Realization Base Case Run with 0.90 kW/m line load and 100 year ventilation T/H curves. (DTN: MO0008SPATHS03.001 [151812])	4.6-7 4.6-8 4.6-9 4.6-10	MO0008MWDBARRI.000 [152184]
Inventory Sensitivity					
SR00_162nm5	Inventory Sensitivity 100 rlz, 1e5 years.	Nominal	Modified Base Case Run to include “new” inventory to address impact of Navy Fuels. DOE-owned SNF Inventory was modified and number of CSNF packages increased.	N/A	MO0008MWDBARRI.000 [152184]

APPENDIX H

SUMMARY AND RESPONSE TO REVIEW COMMENTS ON PREVIOUS YUCCA MOUNTAIN TSPA ITERATIONS

APPENDIX H

SUMMARY AND RESPONSE TO REVIEW COMMENTS ON PREVIOUS YUCCA MOUNTAIN TSPA ITERATIONS

As the science and engineering associated with a potential repository at the Yucca Mountain site has evolved, so too have the analysis capability and the communication of the analyses, models and performance projections. At each stage of the investigation of the repository performance, which corresponds to each iteration of the total system performance assessment (TSPA), internal and external reviews have been performed. These reviews have identified various strengths and weaknesses in the approach, methodology, models, assumptions, data, as well as the documentation of the information in a clear and transparent fashion.

The most recently completed TSPA, TSPA-VA (DOE 1998 [100550], Volume 3) benefitted from a number of reviews, including: the TSPA-VA Peer Review Panel (see Budnitz et al. 1999 [102726] and CRWMS M&O 1999 [153111]), the NRC (Paperiello 1999 [146561]), Clark County, NV (Cohen 1999 [151783]), and the U.S. Geological Survey (Anderson et al. 1998 [101656]). In addition, several Nuclear Waste Technical Review Board (NWTRB) reports and letters have documented issues and concerns they have raised in their reviews of Yucca Mountain project reports.

This Appendix presents in tabular format a summary of the major comments on the TSPA-VA model components and how these components have been updated to address these issues. This summary follows as Table H-1. In order to more traceably identify the component model, this table has been formatted in the same fashion as several other summary-type table, wherein the first column in the key attribute of the repository system and the second column is process model factor or model component of the TSPA-SR model.

While it is difficult to enumerate every comment on the TSPA-VA, Table H-1 presents a summary of many of the most significant comments and how they have been addressed in the TSPA-SR. However, this comment resolution correlation matrix does not completely represent the breadth of comments received on all aspects of the TSPA-VA. For example, numerous issues associated with the models included in the TSPA-VA have been identified in the NRC's Key Technical Issues Issue Resolution Status Reports (IRSRs) as Acceptance Criteria for evaluating the sufficiency of DOE's site recommendation. These issues are addressed in the individual process model reports (PMRs) that most closely correlate to the corresponding key technical issues. Further information on the response to the Peer Review Panel final report is contained in Budnitz et al. 1999 [102726].

While the models and analyses that support the TSPA-SR may not have addressed every issue or comment raised on the previous TSPA iterations, they have addressed the most significant issues. For example, a significant cross-cutting issue raised on the TSPA-VA was the reliance on expert elicitation in the absence of sufficient site or engineering data. In the present analysis there have been no elicitations used in lieu of site data, with the exception of defining the probability of disruptive events.

A common comment raised by the Peer Review Panel and the NWTRB has been the treatment of uncertainties in the analyses and models. The Peer Review Panel noted three broad ways of treating uncertainty, which included (1) conducting additional testing and analyses to update the models and reduce the uncertainty, (2) modifying the design to mitigate the significance of the uncertainty on repository performance or (3) simplify and bound the models or analyses to accommodate the uncertainty without requiring a complete quantitative treatment of the uncertainty. All three of these methods have been used in the TSPA-SR. The models have been significantly updated since the VA The design has been significantly modified (notably to reduce the thermal load, place the corrosion-resistant metal on the outside of the waste package, and add a drip shield above the waste package). Finally, in areas of significant complexity, bounding conservative representations (as summarized in Appendix F) have been employed.

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Limited Water Contacting Waste Package	Climate	Various climatologic, geologic, and hydrologic evidence suggests that the VA's climate models are too conservative.	The climate model remains reasonably conservative but has been improved using additional climatologic evidence. Temporal variability is included as a sequence of climate states. To capture uncertainty, upper-bound and lower-bound climate analogs were chosen for each climate state.	USGS	3.2
	Net Infiltration	Various climatologic, geologic, and hydrologic evidence suggests infiltration/percolation rates are too conservative.	The net infiltration rates remain conservative because of the use of reasonably conservative values for uncertain input parameters. Net infiltration is determined by use of probabilistic simulations that incorporate surface topography and geology. Key uncertain input parameters include precipitation, bedrock and soil hydrologic properties, and evapotranspiration.	USGS	3.2
	UZ Flow	The TSPA-VA UZ flow model possesses uncertainties that have been applied without recognition of their potential limitations. These limitations could call into the usefulness of the sensitivity and uncertainty analyses that have been conducted.	The understanding of the uncertainties in the UZ flow model and their presentation have been improved. Potential limitations have been recognized, and the sensitivity and uncertainty analyses account for them.	TSPA-VA Peer Review Panel	3.2, 5.2
		Occurrence of "bomb-pulse" Cl-36 in Exploratory Studies Facility and implications for groundwater travel times	The occurrence of fast flow paths is implicitly included in the UZ flow model used in TSPA-SR.	State of Nevada	
	Coupled Effects on UZ Flow	The assumption that the fully coupled thermal-hydraulic-mechanical-chemical interactions are of secondary importance needs to be appropriately justified.	Analyses have been conducted to evaluate the effects of chemical and mechanical effects on large scale UZ flow. Based on the results of these analyses, these effects will be insignificant and therefore are neglected.	TSPA-VA Peer Review Panel	3.3

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Limited Water Contacting Waste Package (Cont'd)	Seepage into Emplacement Drifts	The Total System Performance Assessment-Viability Assessment (TSPA-VA) has not fully addressed the potential effects associated with the chemical and mechanical interactions in the thermo-hydrologic analyses.	Thermal-hydrologic-mechanical effects on drift-scale flow have not been fully evaluated but estimates indicate that these effects are likely to be of little importance and are neglected in TSPA simulations. A fully coupled thermal-hydrologic-chemical model has been developed to evaluate changes in hydrologic properties caused by mineral dissolution and precipitation. Predicted changes in hydrologic properties are insignificant and thus neglected in TSPA simulations.	TSPA-VA Peer Review Panel	3.3
		In the TSPA base case, seepage into the drifts is decoupled from Thermal Hydrology, and is assumed to only take place following the end of the thermal period and under ambient flow conditions. These assumptions may be invalid.	Near-drift transients caused by potential repository heating are included in the seepage extraction for TSPA by taking percolation flux above the drifts from the thermal hydrologic model as input rather than simply using the percolation flux from the mountain-scale flow model.	TSPA-VA Peer Review Panel	
	Coupled Effects on Seepage	The effects of the geometrical changes that may result from the collapse of the drift roof in response to thermo-mechanical or seismic processes need to be analyzed.	The seepage abstraction includes effects of drift degradation and flow focussing above the drifts.	TSPA-VA Peer Review Panel	3.3
		Flow in the UZ and thermal-mechanical-hydrologic-chemical interactions must be closely correlated to incoming-water flux.	Thermal-mechanical and thermal-chemical effects on seepage are estimated to be insignificant and thus neglected in the TSPA. Changes in drift geometry are accounted for as is near drift percolation transients caused by repository heating.	TSPA-VA Peer Review Panel	
Long Waste Package Lifetime	In-Drift Physical and Chemical Environments	The data and models used in the VA to calculate the quantity and chemistry of water dripping on waste packages are inadequate to describe process and extent of potential dripping under ambient and thermally-altered conditions.	A new fully coupled thermal-hydrologic-chemical model has been developed to provide compositions of water flowing into the drifts. These compositions are used as input to the new In-drift Salts/Precipitates model. This model uses a Pitzer approach and has been developed to predict the composition of water on the waste packages and in the invert.	NRC	3.3

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Long Waste Package Lifetime (Cont'd)	In-Drift Physical and Chemical Environments (Cont'd)	Not all of the incoming gas will remain in drift or be consumed by in-drift reactions. The cumulative outgoing flux of each constituent gas in the drift should also be taken into account in order to determine the in-drift gas fugacities.	Incoming gas compositions are calculated by the fully coupled thermal-hydrologic-chemical model and are used as input to the in-drift chemistry calculations. Sources and sinks of gas constituents within the drift are neglected based on an analysis that concluded that they are insignificant.	TSPA-VA Peer Review Panel	3.3
	In-Drift Moisture Distribution	When water finally enters the drift, the arbitrarily chosen percentage of evaporated water (90 percent) may give rise to uncertainties in the modeled compositions.	The approach of assuming a percentage of evaporated water was discarded. A new In-drift Salts/Precipitates model was developed using a Pitzer approach to alleviate the difficulties with calculating aqueous and solid compositions for evaporative concentrated solutions. Also, to account for uncertainty in moisture distribution on the waste package, the entire waste package is assumed to be wet and general corrosion is initiated once a critical relative humidity corresponding to the deliquescence point of $NaNO3$ is reached.	TSPA-VA Peer Review Panel	3.4
	Drip Shield Degradation and Performance	N/A	Drip Shield was not a component of the VA design.		3.4
	Waste Package Degradation and Performance	The technical basis for the degradation characteristics and rates of the candidate waste package materials needs to be adequately justified.	Degradation characteristics and rates are based on experimental data from the Project's long-term Corrosion Testing Facility.	NRC	3.4, 5.2
		The TSPA-VA has not fully addressed the potential effects associated with degradation of the drift with time and the effects this may have on waste package performance.	Drift degradation analyses and drip shield structural analyses indicate that effects of rockfall on drip shield performance, WP performance, and thermal hydrology are negligible for TSPA calculations.	TSPA-VA Peer Review Panel	3.3, 3.4

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Slow Rate of Radionuclide Mobilization and Release from the EBS	Radionuclide Inventory and Distribution in Repository	The approach used for screening out unimportant radionuclides from the TSPA analyses needs to be well documented and the potential impacts of not considering the full suite of possible radionuclides present needs to be demonstrated.	A systematic radioisotope screening procedure was conducted and documented that considered the following factors: relative contribution to annual dose, radionuclide longevity, elemental solubility, transport affinity, release scenario, and containment time.	TSPA-VA Peer Review Panel	3.5
	In-Package Environments	There is a need for better definition of the composition of the water that seeps into the waste package and a determination of how this chemistry is modified by reaction with the waste package.	A new model has been developed to predict the evolution of fluid chemistry inside a failed waste package. This information is used by several waste form subcomponent models that are dependent on in-package chemistry.	NRC	3.5
	Cladding Degradation and Performance	Zircaloy may be susceptible to corrosion under certain chemical conditions in the waste package. These chemical processes were explicitly not considered in the TSPA-VA.	Localized corrosion due to fluoride ions in the waste package are considered. Other mechanisms for cladding perforation such as hydride failures, hydride embrittlement, delayed hydride cracking, microbial corrosion, acid corrosion from radiolysis, and enhanced corrosion due to high silica content in seepage waters have been screened out.	TSPA-VA Peer Review Panel	3.5
		The DOE appears to have ignored the judgements of its own experts by taking full cladding credit in the TSPA-VA.	A cladding model is implemented in the TSPA. This model determines the fraction of fuel rods in the commercial spent nuclear fuel packages with perforated cladding as a function of various failure mechanisms induced by physical and chemical processes.	Nuclear Waste Technical Review Board	
	Commercial Spent Nuclear Fuel Degradation and Performance	The presentation of the spent nuclear fuel corrosion models and these side-calculations leaves much to be desired in clarity and transparency. It is difficult to clearly identify when models or codes are used only for side-calculations *versus* their inclusion in the TSPA-VA.	The presentation, technical basis, and documentation of spent nuclear fuel corrosion models are much more transparent and improved. These improvements are, in part, a result of Analysis and Model Reports and Process Model Reports that provide technical bases for TSPA abstraction models and synthesize results, respectively.	TSPA-VA Peer Review Panel	3.5

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Slow Rate of Radionuclide Mobilization and Release from the EBS (Cont'd)	DOE-Owned Spent Nuclear Fuel Degradation and Performance	The presentation of the spent nuclear fuel corrosion models and these side-calculations leaves much to be desired in clarity and transparency. It is difficult to clearly identify when models or codes are used only for side-calculations *versus* their inclusion in the TSPA-VA.	The presentation, technical basis, and documentation of spent nuclear fuel corrosion models are much more transparent and improved. These improvements are, in part, a result of Analysis and Model Reports and Process Model Reports that provide technical bases for TSPA abstraction models and synthesize results, respectively.	TSPA-VA Peer Review Panel	3.5
	Defense High-Level Radioactive Waste Degradation and Performance	Confidence in the extrapolated behavior of corroding borosilicate glass would be greatly enhanced if a mechanism can be identified for this long-term process. Such knowledge could provide the basis for using bounding calculations for glass corrosion rates.	The high-level radioactive waste degradation model is implemented in the form of an analytical expression containing parameters that account for pH, temperature, surface area, glass composition, and solution composition. This expression is widely accepted and used in the scientific literature. Conservative estimates of these parameters are derived from laboratory data to bound long-term degradation rates.	TSPA-VA Peer Review Panel	3.5
	Dissolved Radionuclide Concentrations	The neptunium-bearing phases that control the solution compositions should be identified in the laboratory experiments, and a thoughtful case should be developed for the assumption that this phase will control Np-concentrations in the potential repository environment.	The commercial spent nuclear fuel dissolution model is based on fitting data from flow-through experiments over a wide range of conditions relevant to the repository. Long-term drip testing of commercial spent nuclear fuel has been done over the past six years to determine the relationship between commercial spent nuclear fuel alteration and the release of radionuclides. The model adequately bounds the spread of values reflected in the available dissolution rate data.	TSPA-VA Peer Review Panel	3.5

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Slow Rate of Radionuclide Mobilization and Release from the EBS (Cont'd)	Colloid-Associated Radionuclide Concentrations	The mechanistic basis for the partitioning coefficient for plutonium sorption onto colloids, as well as the stability and filtration characteristics of these colloids, needs to be better defined.	TSPA-SR includes two models for colloid-facilitated transport of radionuclides: one for radionuclides irreversibly bound to colloids, and one for radionuclides reversibly bound to colloids. The irreversible-colloid model is based on data from glass waste-form dissolution tests and colloid-transport tests conducted at the C-wells complex that involved microsphere surrogates for colloids. The C-wells tests were used to define the colloid filtration process, which was modeled as a retardation factor. The reversible-colloid model is based on data from waste-form dissolution tests for commercial spent fuel and estimates of sorption of radionuclides onto similar clays, because measurements of sorption of radionuclides onto colloids could not be completed in time. In order not to underestimate the affect of colloid-facilitated transport, the reversible-colloid model was simplified to use parameters that tended to produce more association of radionuclides to colloids; e.g., the sorption potential for americium onto clay, the largest sorption coefficient of those considered, was used for all radionuclides considered in the model.	NRC	3.5 3.8
	In-Package Radionuclide Transport	The TSPA-VA treatment of the movement of water into a damaged waste package and the transport of radionuclides from such a package were highly conservative.	The models for water movement into damaged waste packages and the subsequent transport and release of radionuclides are conservative and bounding to account for uncertainties in-package transport processes.	TSPA-VA Peer Review Panel	3.6
		In the TSPA-VA transport of corrosion products from the packages through waste package penetrations were judged not to provide any significant retardation to radionuclide releases. The Panel does not accept this view; we believe that it would have been more realistic to have assumed that the resulting penetrations will likely retard radionuclide releases from the waste packages.	There are potentially large uncertainties associated with radionuclide transport and retardation due to sorption onto waste package corrosion products. Because of these uncertainties, retardation due to corrosion products is neglected.	TSPA-VA Peer Review Panel	

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Slow Rate of Radionuclide Mobilization and Release from the EBS (Cont'd)	EBS (Invert) Degradation and Performance	A better definition of the pathways by which water from the corroded spent fuel may escape from the EBS is needed.	The seepage in the drift is conceptualized to flow from discrete fractures above the roof of the drift, falling vertically downward onto the drip shield, the invert, and the waste package if the drip shield has been breached. A detailed description of these flow pathways are provided in the EBS Flow Abstraction.	TSPA-VA Peer Review Panel	3.6
Long Transport away from the EBS	UZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion)	Estimation of a reliable value for the matrix-fracture reduction factor, and validation of the assumptions regarding by-passing of the water around potentially sorbing formations, will add considerable confidence to the validity of the TSPA-VA UZ transport projections.	In general, fractures are modeled as a highly permeable continuum having a low porosity, and the matrix is modeled as a much less permeable continuum having a higher porosity. In the TSPA, fracture-matrix interaction is better represented with an "active-fracture" model, in which only a portion of the fractures is actively flowing under unsaturated conditions.	TSPA-VA Peer Review Panel	3.2, 3.7
		Extrapolation of Busted Butte results to potential repository is questionable.	Experiments conducted at Busted Butte are only used in TSPA-SR to support the results of the modeling, and not directly in the modeling itself. For instance, the imbibition of water into the nonwelded matrix seen at Busted Butte corroborates what is seen in the UZ flow model. Also, the high sorption coefficients measured in samples taken from Busted Butte for several radionuclides tends to justify as conservative the lower sorption coefficients used in the TSPA modeling.	Nuclear Waste Technical Review Board	3.2
	Coupled Effects on UZ Radionuclide Transport	The complexity of transport and sorption, particularly due to the interaction of the UZ with the near-field environment altered zone, requires considerably more work in order to reduce uncertainties to an acceptable level.	In TSPA simulations, the sorption characteristics of the rock are taken to be constant in time. Changes in sorption (or other transport properties) brought about by potential repository induced thermal effects have been considered and found to be insignificant.	TSPA-VA Peer Review Panel	3.7

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Long Transport away from the EBS (Cont'd)	SZ Radionuclide Transport (Advective Pathways; Retardation; Dispersion)	The TSPA-VA has not fully addressed the potential effects associated with dispersion and dilution of radionuclides in the groundwater especially at early times when small source areas may be more likely.	Three modifications have been made to TSPA-SR to address this issue. First, all radionuclides released from the repository during a simulation are concentrated into point sources in the SZ, thereby reducing the effect of dispersion and dilution of radionuclides in the UZ. Second, the particle-tracking method is used in the transport calculations for the SZ, thereby reducing the effect of dispersion and dilution of radionuclides in the SZ. Finally, all radionuclides traveling through the SZ in a simulation are dissolved in the volume of water that is estimated to be used by a farming community of 100 people, as specified in the proposed 10 CFR Part 63 (64 FR 8640), thus effectively eliminating dispersion and dilution as a modeling issue.	TSPA-VA Peer Review Panel	3.8 3.9
		The TSPA-VA lacks site-specific data for the SZ from Forty Mile Wash to the Amargosa Valley.	Nye County has drilled 3 new drill holes in the vicinity of this region. YMP is incorporating data from these drill holes as they become available. TSPA-SR incorporates sorption-coefficient data (in the transport calculations for the SZ), stratigraphic data (as input to the three-dimensional SZ flow and transport model), and water-level data (to calibrate the three-dimensional SZ flow and transport model) from these drill holes.	TSPA-VA Peer Review Panel	
	Wellhead Dilution	No comments on wellhead dilution were made.	Dilution at the well head is conservatively ignored. In the biosphere model, all radionuclides reaching the farming community in groundwater are assumed to be mixed in the volume of water that the community uses.		3.9

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Long Transport away from the EBS (Cont'd)	Biosphere Dose Conversion Factors	The TSPA-VA lacks site-specific data for soil properties for the determination of sorption characteristics.	TSPA-SR includes a description of buildup of radionuclides in the soil. Part of this description involves a leaching coefficient for each radionuclide based on the sorption characteristics of the radionuclide in soil. No sorption data are available for Amargosa Valley soils, so sorption data from similar soil types taken from international references were used in the calculations. Sorption data are available for some radionuclides in alluvium, based on measurements from samples taken from Nye County drill holes, and these data show lower sorption coefficients than those used in the soil-buildup model. Thus, the sorption coefficients used in the soil-buildup model could be conservative (i.e., allow more buildup of radionuclides in the soil). In any event, the TSPA calculations showed soil-buildup to have a small effect on the biosphere dose conversion factors.	TSPA-VA Peer Review Panel	3.9
Minimal Effects of Potentially Disruptive Processes and Events	Probability of Volcanic Eruption	The Panel believes that too much attention may have been devoted to the potential consequences associated with low probability disruptive events such as volcanic events.	The right level of attention has been given to this topic. Although the probability of recurrence is very low during the next 10,000 years, it cannot be ruled out. The conceptual model for igneous activity has three necessary components: 1) a review of the history of past igneous activity in the Yucca Mountain region, 2) development of an estimate of the likelihood of future igneous activity at the proposed repository site, and 3) an analysis of the possible characteristics of a future eruption at the site.	TSPA-VA Peer Review Panel	3.10

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Minimal Effects of Potentially Disruptive Processes and Events (Cont'd)	Characteristics of Volcanic Eruption	DOE's analyses are based on assumptions of physical conditions that are not representative of Yucca Mountain basaltic volcanism.	Information used in the TSPA model to characterize the eruptive and intrusive processes comes from three relevant sources: examination of the geologic record of past intrusive and eruptive events in the Yucca Mountain region; observations of eruptive processes during analogous modern volcanic events elsewhere in the world; and consideration of the range of physical processes that might occur during the interaction between the potential repository and an igneous dike.	NRC	3.10
	Effects of Volcanic Eruption	DOE assumptions about volcanism are not representative of Yucca Mountain; TSPA analyses may underestimate the contribution to risk associated with future igneous activity.	Assumptions are representative: DOE's conceptual model and analyses of volcanism in the Yucca Mountain region are based on expert elicitation from *Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada* (CRWMS M&O 1996 [100116]. This expert elicitation followed NRC guidelines (Kotra et al. 1996 [100909]). Contribution to risk is not underestimated: Expert elicitation produced a probability range of intersection of the repository by a dike, and a conceptual model of the volcanic regime in the Yucca Mountain geologic setting. The TSPA analyses incorporated that conceptual model and the volcanic hazard probability distribution produced by the experts. The current TSPA-SR model for igneous activity provides a reasonable to conservative estimation of the potential risks associated with an igneous event.	NRC	3.10

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Minimal Effects of Potentially Disruptive Processes and Events (Cont'd)	Atmospheric Transport of Volcanic Eruption	DOE assumptions about volcanism are not representative of Yucca Mountain; TSPA analyses may underestimate the contribution to risk associated with future igneous activity.	Contribution to risk is not underestimated: Expert elicitation produced a probability range of intersection of the repository by a dike, and a conceptual model of the volcanic regime in the Yucca Mountain geologic setting. The TSPA analyses incorporated that conceptual model and the volcanic hazard probability distribution produced by the experts. The transport of radioactive material following an eruption is modeled using reasonably conservative assumptions for input parameters. The quantity of ash deposited at any specified point is a function of wind speed and direction, the volume of ash erupted, and the ash particle diameter.	NRC	3.10
	Biosphere Dose Conversion for Volcanic Eruption	DOE assumptions about volcanism are not representative of Yucca Mountain; TSPA analyses may underestimate the contribution to risk associated with future igneous activity.	Contribution to risk is not underestimated: Expert elicitation produced a probability range of intersection of the repository by a dike, and a conceptual model of the volcanic regime in the Yucca Mountain geologic setting. The TSPA analyses incorporated that conceptual model and the volcanic hazard probability distribution produced by the experts. Furthermore, the biosphere modeling for the volcanic direct-release mechanism involves explicit construction of the biosphere dose conversion factors. These biosphere dose conversion factors are based on reasonably conservative assumptions and are in the form of discrete cumulative probability distributions that are used in the TSPA model.	NRC	3.10

Table H-1. Correlation between Principal Factors, Review Comments, and their Resolution in the TSPA-SR (Continued)

Key Attributes of System	Factor	Issue Description	TSPA-SR Response to Issue	Identified By:	Section of TSPA-SR
Minimal Effects of Potentially Disruptive Processes and Events (Cont'd)	Probability of Igneous Intrusion	The Panel believes that too much attention may have been devoted to the potential consequences associated with low probability disruptive events such as volcanic events.	The right level of attention has been given to this topic. Although the probability of recurrence is very low during the next 10,000 years, it cannot be ruled out. The conceptual model for igneous activity has three necessary components: 1) a review of the history of past igneous activity in the Yucca Mountain region, 2) development of an estimate of the likelihood of future igneous activity at the proposed repository site, and 3) an analysis of the possible characteristics of a future eruption at the site.	TSPA-VA Peer Review Panel	3.10
	Characteristics of Igneous Intrusion	DOE's analyses are based on assumptions of physical conditions that are not representative of Yucca Mountain basaltic volcanism.	Information used in the TSPA model to characterize the eruptive and intrusive processes comes from three relevant sources: examination of the geologic record of past intrusive and eruptive events in the Yucca Mountain region; observations of eruptive processes during analogous modern volcanic events elsewhere in the world; and consideration of the range of physical processes that might occur during the interaction between the repository and an igneous dike.	NRC	3.10
	Effects of Igneous Intrusion	The effectiveness of engineered barriers in the event of volcanic activity need to be evaluated.	The TSPA-SR has taken a much more conservative approach to the engineered barriers response during an igneous event than was done in the VA.	NRC	3.10