REPOSITORY SAFETY STRATEGY: U.S. DEPARTMENT OF ENERGY'S STRATEGY TO PROTECT PUBLIC HEALTH AND SAFETY AFTER CLOSURE OF A YUCCA MOUNTAIN REPOSITORY

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Repository Safety Strategy: U.S. Department of Energy's Strategy to Protect Public Health and Safety After Closure of a Yucca Mountain Repository

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CHANGE HISTORY

REV. NO.	ICN NO.	EFFECTIVE DATE	DESCRIPTION OF CHANGE
0		July 1996	Initial issue of strategy, preliminary predecisional draft
1		January 1998	Update strategy to reflect new information, revise title
2		December 1998	Further update strategy to reflect latest performance assessment results and address design options and alternatives

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FOREWORD

This document presents the U.S. Department of Energy's updated strategy to protect public health and safety after closure of a Yucca Mountain repository. It summarizes the current information available to support a postclosure safety case and describes the process for completing this case. This document will be updated as new site, design, and performance information dictates, or when regulatory changes provide impetus for rethinking aspects of the strategy.

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SUMMARY

The Repository Safety Strategy explains the roles that the natural and engineered systems are expected to play in achieving the objectives of a potential repository system at Yucca Mountain. These objectives are to contain the radionuclides within the waste packages for thousands of years, and to ensure that annual doses to a person living near the site will be acceptably low. This strategy maintains the key assumption of the *Site Characterization Plan* (Department of Energy [DOE] 1988) strategy that the potential repository level (horizon) will remain unsaturated. Thus, the strategy continues to rely on the advantages of a repository in the unsaturated zone that enables waste packages assisted by other engineered barriers to contain the wastes for thousands of years. As in the *Site Characterization Plan* (DOE 1988) strategy, the natural system from the walls of the underground openings (drifts) to the human environment is expected to limit concentrations of radionuclides released from breached waste packages.

The Repository Safety Strategy is the framework for the integration of site information, repository design, and assessment of postclosure performance to develop the postclosure safety case that will support the site recommendation and the license application decisions. The safety case will depend on a sound understanding of the expected performance of the repository system. Total system performance assessments indicate that there are four attributes of the repository system that are important to postclosure performance:

- Limited water contacting the waste packages
- Long waste package lifetime
- Low rate of release of radionuclides from breached waste packages
- Radionuclide concentration reduction during transport from waste packages.

These attributes are evaluated through the analysis of principal factors affecting expected postclosure performance for each attribute. The principal factors are evaluated in terms of importance to postclosure performance and are used to identify information needs that will enhance the current understanding of the repository system. Because the principal factors and their relative importance to performance may depend on the design, the strategy also includes consideration of design options and alternatives. Design options and alternatives will be used, along with the natural barriers, to define multiple barriers to provide defense in depth and design margin. The strategy considers disruptive processes and events that have the potential to impact the expected performance of the repository system. The strategy also considers that insights from natural and man-made analogs can provide a degree of independent verification of key aspects of postclosure performance of the repository system. Finally, the strategy considers the important role of the performance confirmation plan in the overall effort to ensure safe performance of the repository system. These will be key elements of the postclosure safety case-the set of arguments that will be prepared to support the site recommendation and license application decisions. Underpinning this set of arguments will be an understanding of the performance of the repository system. This Repository Safety Strategy is the framework to define that understanding.

1. INTRODUCTION

The Repository Safety Strategy is the framework for the integration of site information, repository design, and assessment of postclosure performance to complete the postclosure safety case. The postclosure safety case comprises the information that DOE intends to use to provide reasonable assurance that a repository at Yucca Mountain will adequately protect public health and safety and the environment after the repository is permanently closed. This postclosure safety case will rest upon a basis of information about the Yucca Mountain site, the design of the system, and quantitative estimates of the performance of that system. As information about the site has increased, design has evolved, and performance assessments have become increasingly more sophisticated, the basis for the postclosure safety case has significantly improved. Accordingly, the strategy has evolved as the understanding of what is important to postclosure performance has improved.

The original strategy was described in the *Site Characterization Plan* (DOE 1988). The concepts that provided the basis for that initial strategy have not changed. The strategy still relies upon the attributes of the unsaturated zone environment to provide a setting where waste packages and other engineered barriers are expected to prevent the contact of radionuclides in the waste by groundwater for thousands of years. The strategy continues to address the case where waste packages are breached and multiple natural barriers are relied upon to limit radionuclide movement and concentration. A schematic of the repository in the unsaturated zone is shown in Figure 1.

1.1 SUMMARY OF CURRENT INFORMATION

The information base regarding the site and the design is not yet complete. There are still important uncertainties in the assessment of the system and not all design alternatives have yet been evaluated. As information has increased about the site and the system design, the strategy has focused on the key attributes of this system determined to be important to postclosure performance. These key attributes are:

- Limited water contacting the waste packages
- Long waste package lifetime
- Low rate of release of radionuclides from breached waste packages
- Radionuclide concentration reduction during transport from the waste packages.

The previous version of this strategy (DOE 1998) provided specific hypotheses on the key issues for these attributes based on the information available at the time those versions were issued. Since then, work has been done to evaluate those hypotheses both in terms of testing and analyses and performance assessments based on previous work. This version of the Repository Safety Strategy documents a summary of what is now known about each of the hypotheses and identifies nineteen principal factors. These principal factors are those factors that bear directly or indirectly on one or more of the components of the repository system identified by sensitivity

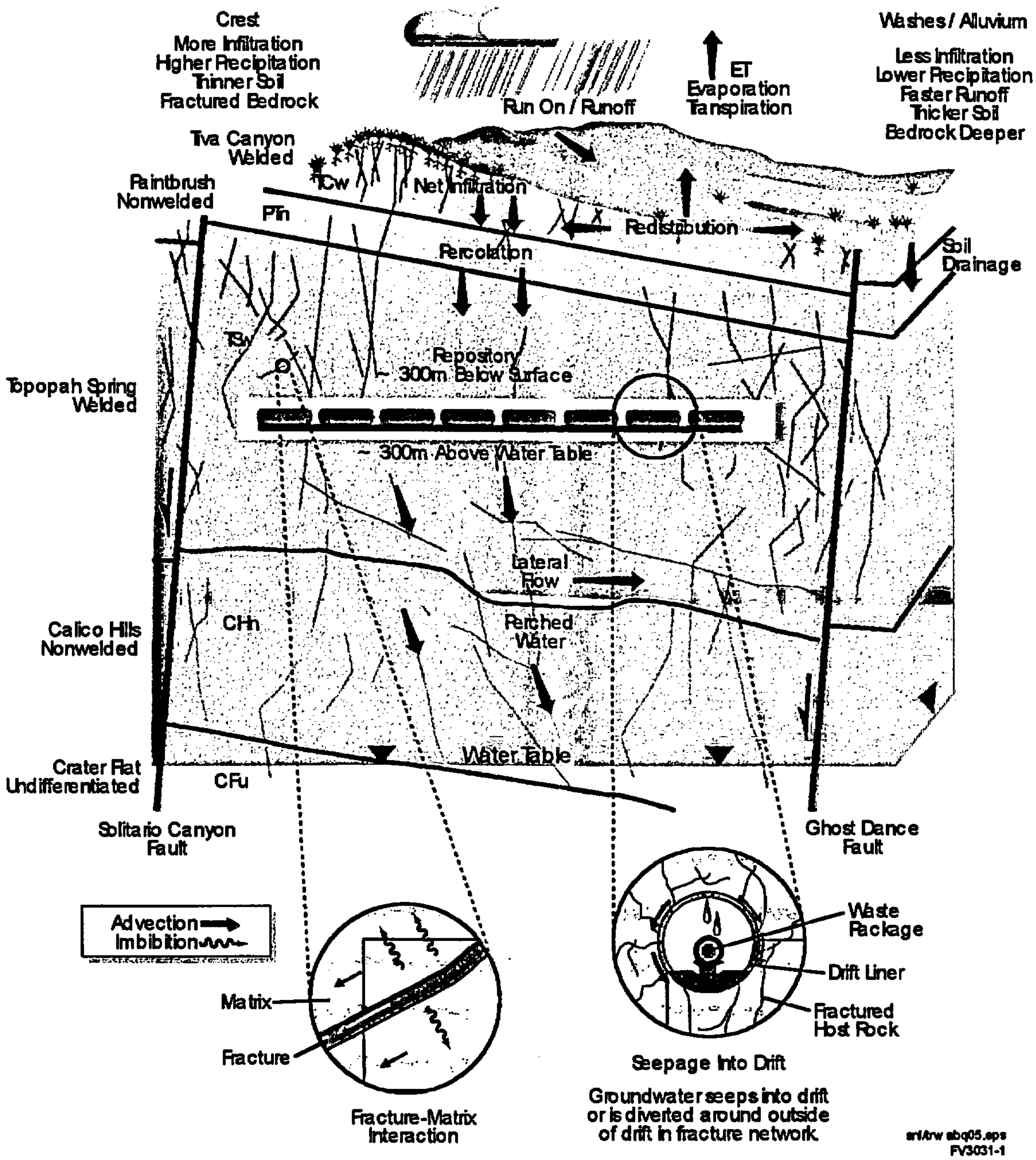


Figure 1. Schematic of a Repository in the Unsaturated Zone at Yucca Mountain

studies as being potentially important to understanding system performance. The principal factors replace the testing hypotheses as the framework for focusing the remaining work for site recommendation and license application. This revision of the strategy describes those factors and provides an assessment of their relative importance. This strategy also provides information about potential design options and alternatives since the principal factors and their relative importance may depend on the design features selected.

Before describing the principal factors and ongoing work to evaluate design options and alternatives, a summary is provided of what is known about the system, focusing on the key attributes and what has been learned from evaluating the hypotheses specified in the previous version of this strategy (DOE 1998). The following sections discuss the key attributes, the hypotheses to address their associated issues, the conclusions drawn about those hypotheses, the identification of related principal factors, and design options considered in evaluating the VA reference design. Appendix A provides an overview of the VA reference design.

1.1.1 Limited Water Contacting the Waste Packages

Performance assessments have shown that the amount of water contacting the waste packages is the most important determinant of the ability of the site to contain and isolate waste. The amount of water contacting waste packages ultimately affects all aspects of performance from waste package lifetime to radionuclide movement. The amount of water contacting the waste packages is limited by the seepage into the repository. The seepage into the repository depends on the nature of percolation in the repository host rock that depends, in turn, on precipitation at the surface, the amount of this precipitation infiltrating into the mountain, the properties of the subsurface rock, and the redistribution of the water as it percolates down to the host rock.

The hypotheses specified for this attribute are:

1. Percolation flux at repository depth can be bounded.
2. Seepage into drifts will be a fraction of percolation flux.
3. Thermally induced seepage can be bounded.
4. Seepage that contacts waste packages can be limited.

The work to address these hypotheses has provided additional information about the repository system. That information points to the identification of principal factors and some design options for the VA reference design potentially important to postclosure performance. The hypotheses and the principal factors and design options following from this work are summarized in Table 1. They are discussed below.

Hypothesis 1--Percolation flux at repository depth can be bounded

Investigations to evaluate the bounds to the percolation flux have determined that the current percolation flux at the Yucca Mountain site averages between 2 and 15 mm/year and best estimates give an average of about 7 mm/year over the proposed repository footprint. Climate studies suggest that average precipitation over the next million years may be a factor of two higher than present-day rates, and projections using models calibrated to present day precipitation suggest that average percolation flux over the long term may be on the order of 40 mm/year. This work indeed suggests that the average percolation flux at the site can be bounded. At the same time, the percolation flux is both spatially and temporally variable. The range of this variability is not known at the present time and bounds to the local flux still need to be established. The work has shown that the particular value of percolation flux is critical to the determination of seepage into the emplacement drifts. Further, these studies point to the potentially important role of precipitation, net infiltration into the mountain, and percolation to depth as principal factors important to system performance.

Hypothesis 2--Seepage into drifts will be a fraction of percolation flux

In situ measurements have provided useful information about the nature of the seepage, and modeling studies calibrated to these measurements have provided indications of the fraction of percolation that would contribute to this seepage. The seepage process model has been tested against preliminary data from the liquid release tests in niches in the Exploratory Studies Facility (ESF)[1] (Wang et al. 1997). The model appears to fit the data reasonably well (Chapter 2.4.4.9 and 2.7.4, Civilian Radioactive Waste Management System Management and Operating Contractor [CRWMS M&O], 1998a). Estimates of the seepage have considered spatial and temporal variations in the percolation flux and the properties of the rock and the fluxes that might prevail in the future. These analyses all seem to be consistent with the hypothesis. In addition to this conclusion, the work has provided additional insight into the potential effect of seepage on performance. The work also suggests that the seepage fraction will be highly variable across the repository. Further, because the local percolation flux is difficult to predict, the seepage will also be difficult to forecast. These analyses suggest that seepage into drifts is a principal factor important to system performance.

Hypothesis 3--Thermally induced seepage can be bounded

Equivalent continuum modeling, correlated to measured rock property distributions, suggests limits to the changes to the flow system. Laboratory work and results of *in situ* thermal testing appear to be confirming many of these effects on the local scale. This work continues to support the concept that there will be a period in which the percolation in the vicinity of the drifts and the associated seepage will be diminished or eliminated entirely (Section 4.4.1, CRWMS M&O 1998a). Flow modeling studies suggest that after this period, water may flow back into the dryout zone for a time. It is conceivable that seepage could increase temporarily during this reflux period. At the same time coupled-effects studies suggest that a mineral cap could be formed over the drifts when condensate above these drifts dissolves minerals, begins to flow back, and the dissolved minerals in this refluxing water precipitate under the higher temperatures near the drifts (e.g., calcite solubility decreases with temperature). The additional mineral filling of fractures could limit percolation toward the drifts and seepage into them. Therefore, the theoretical studies suggest seepage into drifts may be limited by thermal perturbations. This

[1] The Exploratory Studies Facility is an 8 km long tunnel at the eastern margin of the proposed repository block.

Table 1. Identification of Principal Factors and Design Options from Previous Work

Hypotheses (Rev 1)	Principal Factors and (Design Options)
1. Percolation flux at repository depth can be bounded	• Precipitation and infiltration into the mountain • Percolation to depth
2. Seepage into drifts will be a fraction of percolation flux	• Seepage into drifts
3. Thermally induced seepage can be bounded	• Effects of heat and excavation on flow
4. Seepage that contacts waste packages can be limited	• Dripping onto waste package • (Drip shield plus backfill)
5. Heat reduces relative humidity at waste package surface	• Humidity and temperature at waste package
6. Corrosion of waste package materials is slow	• Chemistry of water on waste package • Integrity of inner corrosion-resistant waste package barrier
7. Protection of inner barrier by the outer barrier	• Integrity of outer carbon steel waste package barrier
8. Engineered enhancements can extend the containment time	• (Ceramic waste package coating) • (Drip shield plus backfill)
9. Containment time will be sufficient to prevent oxidation of spent fuel	• Integrity of spent nuclear fuel cladding
10. Water that contacts waste can be limited	• Seepage into waste packages
11. Release rate of soluble radionuclides controlled by slow waste form dissolution	• Dissolution of spent nuclear fuel and glass waste forms
12. Release rate of actinides controlled by solubility limits rather than colloidal stability	• Neptunium solubility • Formation and transport of radionuclide-bearing colloids
13. Physical properties of barriers reduce concentrations during transport	• Transport through and out of the engineered barrier system • Transport through the unsaturated zone • Flow and transport in the saturated zone
14. Chemical properties of barriers reduce concentrations during transport	• Transport through and out of the engineered barrier system • Transport through the unsaturated zone • Flow and transport in the saturated zone
15. Contaminants in the lower volume flow in unsaturated zone will be diluted by higher volume flow in the saturated zone	• Flow and transport in the saturated zone • Biosphere transport and uptake

work suggests that the thermally induced seepage may well be bounded, but there is a need to continue study of the effects of heat and excavation on flow. Continuation of the *in situ* thermal testing and measurements of near-field water mobilization and thermomechanical response of the rock, including observations of the effects after the heaters are turned off, will substantially improve the model of the effects of heat on the flow system. This information and monitoring of the conditions in the vicinity of the ESF and the Cross Drift[2] will provide additional information and confidence in the representation of the effects of the excavation on the flow system as well. The work to date suggests that the effects of heat and excavation on the flow system is a principal factor potentially important to postclosure performance.

Hypothesis 4--Seepage that contacts waste packages can be limited

The current performance assessment modeling assumes that any water dripping into the emplacement drifts contacts waste packages (Section 5.10.1, CRWMS M&O 1998a). This assumption provides an upper bound to the amount of water that can drip onto the waste package from fractures. However, this approach does not lead to any refinement of the understanding of this effect. It does not, for example, address formation of moisture films on the drift walls that divert water away from the waste package, nor evaporation and condensation of seepage away from the waste packages. The modeling studies support the view that dripping onto the waste package is a principal factor potentially important to postclosure performance of the repository system. However, until more definitive information is available, estimates of the range of conditions will have to be bounding, and conservative, based on the current information. If seepage enters the drift, engineered enhancements can protect the waste packages from contact with seepage. As discussed in Section 3, analysis of the performance of drip shields and backfill show that these engineered enhancements can limit the amount of seepage that contacts the waste packages.

1.1.2 Long Waste Package Lifetime

As long as waste packages remain intact, the waste will be completely contained and prevented from any contact with the host rock, air, or groundwater. This containment has several positive results. The radiation source is reduced over time due to radioactive decay. Uranium dioxide is protected from contact with air while it is at the higher temperatures that make it more susceptible to oxidation. In addition, the waste is protected during the period of greatest uncertainty about processes operating in the repository, during the initial thermal period.

The hypotheses specified for this attribute are:

5. Heat reduces relative humidity at the waste package surface.
6. Corrosion of waste package materials is slow.
7. The inner waste package barrier will be protected by the outer barrier.
8. Engineered enhancements can extend the containment time.

[2] The Cross Drift is a drift that begins at the northern reach of the ESF and follows a southwesterly trajectory across and above the proposed repository block.

The work to address these hypotheses has provided the following information:

Hypothesis 5--Heat reduces relative humidity at the waste package surface

Observations in the ESF both under ventilated and unventilated conditions have led to high confidence in the estimates of the ambient temperature and humidity at depth, and therefore the conditions that will exist in the emplacement drifts in the long term. The conditions are likely to be different from the ambient conditions in the first few thousand years when temperatures are increased by the waste heat generation. However, the ranges of temperature and humidity during this period have been extensively modeled and compare favorably with the results from *in situ* thermal testing. These results appear to be consistent with this hypothesis. The work continues to support the view that the predictability of the environment within the emplacement drifts is an important factor in the performance of the engineered barrier system. This view is captured by the identification of the humidity and temperature on the waste package as a principal factor potentially important to performance.

Hypothesis 6--Corrosion of the waste package materials is slow

Corrosion tests have shown that corrosion rates for the candidate materials for the outer barrier of the reference dual-barrier waste package design are not slow. In spite of this understanding, it does appear that the outer barrier of the waste package can be designed to provide significant physical protection for the inner barrier during the first few thousand years after closure (see Hypothesis 7 below).

Corrosion testing of candidate corrosion-resistant materials for the inner barrier to address this hypothesis indeed indicates that at low relative humidity, corrosion rates for these materials are very low. Further, the corrosion rate of the corrosion-resistant candidate materials such as Alloy 22 and titanium alloys is negligible at low humidity. The results are so far consistent with this hypothesis for the corrosion-resistant materials.

The work to address this hypothesis has led to an even broader conclusion regarding the corrosion rate of the corrosion-resistant materials. These studies have shown that, even under aggressive conditions (very low pH, high temperature), Alloy 22 has corrosion rates of less than one micron/year. Extrapolation of the corrosion rate to repository conditions leads to large uncertainty (approximately three orders of magnitude) but it is clear that, even under this range of uncertainty, corrosion of this material is likely to be very slow under ranges of conditions likely to occur in the repository.

The associated sensitivity studies show that the integrity of the corrosion-resistant material is highly important to system performance. They also show the importance of the remaining uncertainties. These include the chemical effects on the corrosion rate, particularly in crevices at the surface of the material where pH can be very low. There are some uncertainties in the estimates of the ranges of chemistry that will occur in the repository system, including possible excursions in the chemistry due to heat and rock-water interactions. Further, the chemistry could be affected as seepage contacts other materials in the repository including concrete drift liners and iron components of the engineered barrier system. These effects are likely to be critical for the performance of the waste packages. Therefore, the work to address this hypothesis identifies two principal factors: integrity of inner corrosion-resistant waste package barrier and chemistry of water on waste package.

Hypothesis 7--The inner waste package barrier will be protected by the outer barrier

Originally this hypothesis was proposed because of the possible galvanic protection of the inner barrier by the mild steel of the outer barrier. However, studies have not been able to demonstrate that this effect will be universal. In addition, galvanic coupling between the two barriers may result in potentially adverse effects to long-term performance of the waste package by enhancing corrosion of the outer barrier and by causing hydrogen embrittlement of Alloy 22. These potentially adverse effects will be addresed in future iterations of total system performance assessment.

At the same time, studies of the ability of the outer barrier to protect the inner barrier against rockfall have shown that this barrier will provide significant protection for several thousand years while it maintains its strength. In addition, it appears that the outer barrier would serve to protect the inner barrier from scratches or other effects that could damage the corrosion passivation layers on its surface. Thus, the outer barrier would appear to provide significant physical protection to the inner barrier in the first several thousand years after emplacement.

The associated studies have provided additional information about the role of the outer barrier with respect to postclosure performance. These studies suggest that, although corrosion rates are not low for the candidate materials under consideration for the outer barrier, it nevertheless will play an important role in the postclosure performance of the repository system. The integrity of the outer steel waste package barrier therefore appears to be a principal factor potentially affecting system performance.

Hypothesis 8--Engineered enhancements can extend the containment time

This hypothesis has been addressed by performance assessment sensitivity studies. Although sensitivity studies that examined the performance of enhancements such as drip shields over the waste package and ceramic coatings to the waste package are very preliminary, they show important effects on system performance (Chapter 8, CRWMS M&O, 1998a). The importance arises from the fact that, while the vast majority of radionuclides in the repository are immobile and will not migrate to the accessible environment, a few of the radionuclides are mobile and can migrate if water contacts them. Enhancements that contribute to containment time or otherwise prevent water from contacting the waste could have a significant impact on performance with respect to the mobile radionuclides either directly or by providing defense in depth for the system.

1.1.3 Low Rate of Release of Radionuclides from Breached Waste Packages

The rate of release of radionuclides from breached waste packages is one of the key factors determining the peak dose rate. Most of the radionuclides that would be in the repository are not mobile at the Yucca Mountain site: they are insoluble or they sorb strongly to minerals and materials in the repository and cannot move out of the repository. The current information for these radionuclides is sufficient to complete the postclosure safety case regarding them. A small fraction is relatively mobile, however, and could be transported away from the repository if contacted by water.

Hypotheses established for this attribute are the following:

9. Containment time will be sufficient to prevent oxidation of spent fuel.
10. Water that contacts waste can be limited.
11. The release rate of soluble radionuclides is controlled by slow waste form dissolution.
12. The release rate of actinides is controlled by solubility limits rather than colloidal stability.

Studies to address these hypotheses have provided the following information:

Hypothesis 9—Containment time will be sufficient to prevent oxidation of spent fuel

The issue addressed by this hypothesis is integrity of the spent fuel cladding, a potential factor limiting mobilization of radionuclides in breached waste packages. The concern is that oxidation of the uranium oxide (UO_2) could lead to swelling of the fuel pellets and bursting of the cladding. Extensive testing of the Zircaloy has resulted in a good understanding of its properties and there is high confidence that this material will last under repository conditions and provide significant limits on mobilization of radionuclides if it remains intact. The studies suggest that if the waste packages remain intact and keep oxygen from the waste long enough for temperatures to diminish, the uranium oxide will not oxidize to the higher-volume U_3O_8 state. These results appear to be consistent with this hypothesis. The associated performance assessment sensitivity studies regarding the role of spent fuel cladding have highlighted the overall importance of the principal factor of integrity of spent fuel cladding to performance of the system. The information proceeding from the work to address this hypothesis therefore suggests that spent fuel cladding is a principal factor potentially important to system performance.

Hypothesis 10—Water that contacts waste can be limited

The current models assume either that all the water contacting breached waste packages enters the waste packages or that the amount seeping into the packages is proportional to the area of the waste package that is actually breached. These assumptions bound the range for the flow into the waste package and confirm the hypothesis.

Associated work to study the importance of this assumption have demonstrated the important role of seepage into the waste package with respect to mobilization of solubility-limited radionuclides such as neptunium-237. If small breaches in the waste package would limit the amount seeping by capillary or other effects, performance of the system could be significantly affected. The work therefore suggests that seepage into the waste package is a principal factor potentially important to system performance.

Hypothesis 11--The release rate of soluble radionuclides is controlled by slow waste form dissolution

Studies to address this hypothesis have led to the following. Dissolution rates of both uranium oxide and glass have been measured under laboratory conditions. These measurements confirm that the mobilization rate of soluble radionuclides such as technetium-99 is limited by the waste form dissolution rate.

Associated performance assessment studies indicate that the soluble radionuclides, technetium-99 and iodine-129, are among the most important contributors to system performance. The work to address this hypothesis therefore suggest that the rate of degradation of the spent fuel or glass waste form are critical to system performance as manifested in the release of these radionuclides. This information therefore points to dissolution of spent nuclear fuel and glass waste forms as a principal factor potentially affecting postclosure system performance.

Hypothesis 12--The release rate of actinides is controlled by solubility limits rather than colloidal stability

The hypothesis focuses on the potential to mobilize radionuclides by the formation of radionuclide-bearing colloids in the repository or the attachment of radionuclides to colloids in the groundwater. Studies have shown that colloids exist in the waters at Yucca Mountain and laboratory studies and measurements have shown that colloids in the saturated zone at the nearby Nevada Test Site can carry radionuclides generated in atomic testing. Laboratory studies of colloids and modeling based on these data have been used to estimate the potential for colloidally-assisted transport at the site. Field studies using microsphere surrogates for colloids suggest that filtration mechanisms could limit the effectiveness of transport by colloids. The laboratory studies suggest that under some conditions desorption of radionuclides from the colloids will occur and the overall effect in the field could be minor; however, the work is not yet sufficient to confirm that this is the case at Yucca Mountain. Because of the potential importance to assessing system performance, the formation and transport of radionuclide-bearing colloids has been identified as a principal factor.

The work to address this hypothesis has led to important insights about the factors affecting performance. Laboratory studies have confirmed that the solubility limits of virtually all actinides are sufficiently low that, barring formation of or attachment to colloids, fractional release rates are likely to be much lower than the waste form dissolution rate. A key question is the solubility limit for neptunium. The dominant stable and metastable phases controlling the dissolution of neptunium have been investigated in both experiments and modeling and there is high confidence that the neptunium solubility considered in the performance assessments represents an upper bound to this factor. Further, performance assessment sensitivity studies indicate that neptunium-237 is a key contributor to long-term performance of the system; consequently, neptunium solubility also appears to be a principal factor for assessing postclosure performance.

1.1.4 Radionuclide Concentration Reduction During Transport from the Waste Packages

Radionuclides that are released from the breached waste packages must migrate through the engineered barrier system and enter the unsaturated-zone flow system in the host rock before they eventually reach the aquifers beneath the site. Potential dose rates through the unsaturated-zone and the saturated zone flow systems can be reduced during this transport. The dose rate depends directly on the concentration of radionuclides in the water. These concentrations change as the radionuclides migrate from the repository to the point of potential uptake by individuals using the water. In general, heterogeneities in the flow and transport properties cause dispersion; precipitation, matrix diffusion, and sorption cause depletion. Both dispersion and depletion cause reduction of the radionuclide concentrations. The biosphere is a principal factor in determining dose. It is considered an important factor that must be understood. It is not considered a barrier in the same sense as the other principal factors are so considered.
Hypotheses established for this attribute are the following:

13. Physical properties of barriers reduce concentrations during transport.
14. Chemical properties of barriers reduce concentrations during transport.
15. Contaminants in the unsaturated zone will be diluted by higher volume flow in the saturated zone.

The studies to address these hypotheses have led to the following information:

Hypothesis 13--Physical properties of barriers reduce concentrations during transport
Hypothesis 14--Chemical properties of barriers reduce concentrations during transport

Studies to test these hypotheses have examined the role of engineered and natural barriers in retarding migration of radionuclides and in reducing radionuclide concentration in groundwater flowing through them. There are few data regarding transport within waste packages; however, there are data for diffusion of radionuclides through granular material and data regarding sorption onto iron corrosion products that may be outside the waste packages. These analyses show some effect on concentrations. Studies show that the natural barriers (e.g., the host rock and the rock underlying the repository) may have some effect on the concentrations. Although the variability of the rock properties precludes definitive statements about the precise effect, the laboratory data strongly suggest that the concentrations will be reduced during migration through this rock. The preliminary data suggest that concentrations of those radionuclides that sorb onto the minerals in the rock will be significantly reduced during transport, but that those that do not sorb readily (such as iodine and technetium isotopes) may not be significantly affected. These results appear to be consistent with the hypothesis for most radionuclides, but suggest that the hypothesis is not valid for these mobile radionuclides. These studies support the view that transport through and out of the waste package, transport through the unsaturated zone, and flow and transport in the unsaturated zone are principal factors potentially important to postclosure performance.

Hypothesis 15—Contaminants in the unsaturated zone will be diluted by higher volume flow in the saturated zone

The dilution factor in the saturated zone is a sensitive parameter for system performance, but the appropriate range of uncertainty for this factor is controversial (Chapter 8, CRWMS M&O 1998a). A data gap exists along the flow path from the repository from 5 to 20 km; this is a relatively large section of the potential radionuclide transport pathway. For the base case for the current total system performance assessment, the aggregate distribution for the dilution factor developed by an expert elicitation panel was used to reduce the maximum concentration in the ground water approximately 20 km down gradient from the repository (Chapter 8, CRMWS M&O, 1998a). The studies indicate that average flux in the saturated zone, a key factor for the estimate of dilution, is not strongly sensitive to model assumptions. However, local variations can be quite high: in particular the models are consistent with locally fast pathways involving faults and channeling in fractures. Effective porosities, conductivities, and dispersivities also needed for estimates of dilution are not well established at present, with orders-of-magnitude differences between single-hole and cross-hole tests. The degree of mixing of waters from different sources is not known. It is possible that there may be mixing of waters across subbasins, and information from one well suggests that there may be upwelling of water south of the site from the deep aquifers; however, the information is not sufficient to determine whether there is significant mixing. Measurements in the C holes indicate mechanical dispersion on a scale of 30 m is comparable to that measured at most other sites. No information is available to enable extrapolation of these data to the km scale.

This hypothesis has also led to consideration of possible dilution during pumping from aquifers that might contain contaminated water. The dilution will be a function of the ratio of the volumetric pumping rate to the volumetric flow rate of water contacting the waste. Conceivably this could be quite large. However, pumping could also withdraw water from large distances and sample large portions of the contaminated plume. The overall effect is not known at the present time; however, it is clear that any dilution occurring in the saturated zone and during pumping could be important to performance. This information supports the view that flow and transport in the saturated zone and dilution from pumping are principal factors potentially important to the assessment of postclosure performance of the repository system.

1.1.5 Disruptive Processes and Events

The strategy also addresses disruptions to the system that potentially could release radionuclides directly to the human environment or otherwise adversely affect the characteristics of the system. Specific processes and events that have been identified in this regard include tectonics and seismicity, volcanism, human interference, and nuclear criticality. Hypotheses to address the first two of these categories are the following:

16. Fault displacement impacts will not be significant.
17. Ground motion impacts will be minimal.
18. Consequences of volcanism will be limited.

No hypotheses were proposed for human interference and nuclear criticality. With regard to the potential for human interference, the assessments of the Yucca Mountain site and region have suggested that the site is not a likely target for future exploration. Accordingly, no hypotheses regarding human interference have been proposed. No hypothesis is defined for nuclear criticality because the information about the characteristics of the waste form, the corrosion of waste packages, and the dissolution and transport of radionuclides and neutron absorbers needed to evaluate this possibility will become available through evaluation of other hypotheses.

Studies to address the proposed hypotheses have led to the following information:

Hypothesis 16– Fault displacement impacts will not be significant
Hypothesis 17–Ground motion impacts will be minimal

Current information about the tectonic regime at Yucca Mountain suggests an upper bound to both fault movement and mean ground acceleration at the site, confirming these hypotheses. The associated work also shows that quantitative estimates of postclosure performance for these upper bound conditions do not indicate any significant releases for these conditions. This work suggests that these issues can be addressed by current information and no principal factors have been identified.

Hypothesis 18–Consequences of volcanism will be limited

Site characterization work has led experts to conclude that the likelihood of significant future volcanic activity in the vicinity of the site is negligible. Performance assessments that consider several extreme volcanism scenarios do not indicate significant releases of radionuclides. This work suggests that this issue can also be addressed by current information and no principal factors have been identified.

2. DISCUSSION OF THE PRINCIPAL FACTORS IMPORTANT TO POSTCLOSURE PERFORMANCE

Section 1 introduced the principal factors potentially important to postclosure performance of the repository system at the Yucca Mountain site. This section provides a more detailed discussion of the principal factors. These principal factors reflect the current understanding of the site and the performance assessment sensitivity studies that have been conducted for the VA reference design.

The discussion in Section 1 gave a general context for these principal factors in terms of the VA reference design because they depend to some extent on the particular design. That design includes robust waste packages emplaced in drifts about 300 m below the surface and about 300 m above the water table. The drifts are not backfilled but are lined with concrete in order to control the ground during preclosure operations. The specific factors also depend upon the characteristics of the radionuclides that are at issue. These radionuclides fall into three categories. One category is the immobile radionuclides. These radionuclides are so insoluble that, even accounting for the range of uncertainty in their properties, they are unlikely to be released from the repository before they decay to stable isotopes, even if there are no engineered barriers to contain them. This category includes such radionuclides as the thorium isotopes. Beyond their solubilities, there is no factor critical to repository safety. Because these solubilities are sufficiently well established, they are not considered further in this strategy.

A second category includes the radionuclides that may, under some conditions, be mobile at Yucca Mountain if they are released from the repository. The possible ranges of solubilities, retardation factors, and other properties of these radionuclides that are consistent with the current data indicate that they are sufficiently mobile under certain conditions that they could reach the biosphere downgradient from the repository. This category includes the plutonium, actinium, and neptunium isotopes. While the potential hazard of these radionuclides is considerable, features of the site and the engineered barrier system play a critical role in limiting the risk to the public. These features therefore constitute principal factors for the performance of the repository system vis-à-vis these radionuclides. The principal factors include those that affect the amount of water that might contact the waste packages; the lifetime of the waste packages; and the mobilization, transport, and dilution of radionuclides released from breached waste packages.

The third category is for the radionuclides that are highly mobile at this site. These include radionuclides that are highly soluble and that, when dissolved in water, travel as anions, not sorbing to minerals. These radionuclides include iodine and technetium isotopes. The total inventory of these radionuclides in the repository will be very small, less than 0.004% of the total inventory, but sufficient to result in a significant dose rate to individuals if they are released into the ground water and ingested. This category also includes radionuclides that attach to or form as colloids and that readily travel through the system. The total inventory of such radionuclides is uncertain but could include some fraction of the plutonium, europium, cesium, and other isotope inventory. The principal factors in this case are those that affect the contact of water with the waste packages and the lifetime of those waste packages.

Table 2 presents the set of principal factors that have been identified for the VA design. These principal factors capture the key sensitivities identified in the total system performance assessment (TSPA) analyses for the radionuclides in the latter two categories. The following section summarizes each principal factor and its system level sensitivities. Table 2 and the discussion are organized according to the four key attributes of the system:

- Limited water contacting waste packages
- Long waste-package lifetime
- Low rate of release of radionuclides from breached waste packages
- Radionuclide concentration reduction during transport from the waste packages.

2.1 LIMITED WATER CONTACTING WASTE PACKAGES

2.1.1 Precipitation and Net Infiltration into the Mountain

Precipitation (e.g., rainfall, snowfall) and net infiltration are important because they are the source of water that can flow down to the repository horizon. That flow is the source of water that could lead to corrosion of metal components of the waste package and could mobilize radionuclides in breached waste packages. Precipitation and infiltration vary over the surface of Yucca Mountain, and are expected to vary over time (Section 2.2, CRWMS M&O, 1998a). Performance assessments indicate that the mobile radionuclides, although not sensitive to the amount of water, are sensitive to the amount of waste exposed to this water. They also indicate that neptunium-237 is sensitive to both the amount of waste exposed to water and, because it is solubility-limited, the amount of water contacting the waste. Over all, sensitivity analyses indicate that infiltration is important to repository performance because of its influence on percolation flux and seepage into drifts (Section 2.7, CRWMS M&O, 1998a). This principal factor is moderately important to system performance based on the variation in range of plausible infiltration rates. Higher infiltration rates, in general, may adversely impact performance. However, the impact of increased infiltration rates must be considered in conjunction with other components of the repository system, such as the uncertainties in seepage and corrosion of the waste package, to understand the overall impact on performance (Section 2.7, CRWMS M&O, 1998a).

2.1.2 Percolation of Water to Depth

Percolation in the host rock provides the source of water that can seep into the emplacement drifts and contact the waste packages, and that flows from the repository horizon down to the water table. It is the result of interaction between net infiltration at the ground surface, and the flow pathways in the rock above the repository. Percolation flux will vary with time, and with location in the repository. Performance assessment sensitivity studies indicate that this factor is of low importance to performance. This outcome is, however, an artifact of the way the modeling was done, with uncertainty in precipitation and infiltration determining and, thus, dominating the uncertainty in percolation. Therefore, the influence of the factor "percolation to depth" should be similar to that of precipitation and infiltration. The latter two factors provide

Table 2. Principal Factors Affecting Postclosure Performance for the VA Design

Key Attributes of Repository System	Principal Factors
Limited water contacting waste packages	Precipitation and infiltration into the mountain
	Percolation to depth
	Seepage into drifts
	Effects of heat and excavation on flow
	Dripping onto waste package
	Humidity and temperature at waste package
Long waste package lifetime	Chemistry of water on waste package
	Integrity of outer carbon steel waste package barrier
	Integrity of inner corrosion-resistant waste package barrier
Low rate of release of radionuclides from breached waste packages	Seepage into waste package
	Integrity of spent nuclear fuel cladding
	Dissolution of spent nuclear fuel and glass waste forms
	Neptunium solubility
	Formation of radionuclide-bearing colloids
	Transport through and out of the engineered barrier system
Radionuclide concentration reduction during transport from the waste packages	Transport through the unsaturated zone
	Flow and transport in the saturated zone
	Dilution from pumping
	Biosphere transport and uptake

the input values to the percolation model, however the amount and redistribution of water as it travels through the unsaturated zone to the drift is contained in the model for the percolation flux.

Without an adequate characterization of the volume and location of flowing water, it is not possible to defensibly calculate the seepage into the drift. The uncertainty in the peak dose rates depends strongly on the fraction of waste packages contacted by seepage (Section 11.4, CRWMS M&O, 1998a). Therefore, the percolation that determines the seepage is probably at least moderately importance to the peak dose rate.

2.1.3 Seepage into the Emplacement Drifts

Seepage into the emplacement drifts is the principal source of water that may drip onto waste packages, contributing to waste package corrosion and mobilization of radionuclides. The variability of this factor is greater than that for the percolation flux at the repository horizon, because seepage is sensitive to the variable geometry and hydrologic properties of fractures near the drift wall. The hydrologic properties can divert percolation water through the rock, bypassing the drift openings and these properties may be changed in the vicinity of the drifts by the excavation of those drifts or as a result of near-field thermomechanical stresses. The sensitivity studies indicate that this principal factor is important to postclosure performance because it has a direct effect on the number of waste packages that fail, and it has a large uncertainty (Section 2.7, CRWMS M&O 1998a).

2.1.4 Effects of Heat and Excavation on the Flow

Heat will be generated in the repository due to radioactive decay of the waste, increasing temperatures in the emplacement drifts and out in the host rock for a few thousand years, until the waste cools. The heat will redistribute moisture in the host rock, and drive physical and chemical processes that could result in long-term changes to the flow properties of the rock (Chapter 3, CRWMS M&O 1998a). Excavation of the drifts, mechanical loading of the rock mass by thermal stress, and drift collapse could modify the flow properties as well (Chapter 3, CRWMS M&O 1998a).

No specific performance assessment sensitivity studies have been conducted for this factor. The effect of heat is a relatively short-lived phenomenon, but it may have a significant influence on the ambient flow fields. Thermohydrologic studies for Yucca Mountain to date have not determined thermally induced changes in the host rock hydrologic properties. Some analyses have shown that changing the properties could lead to a different dry-out period than is currently calculated. This could potentially change repository system performance. In addition, this factor is expected to influence the percolation flux and seepage into the drift by potentially changing the hydrologic properties of the rock mass by mechanical means resulting in an introduction of more permeability. It could also result in alteration of the rock by chemical means. This could occur either by decreasing permeability by precipitating minerals that close fractures and pores, or by altering the sorptive properties in the rock under the repository. Current analyst judgment is that this factor does not significantly influence the results, given the base case models currently being applied (Section 3.7, CRWMS M&O, 1998a). After further evaluations are completed, this factor is expected to receive a low importance rating if the magnitude of thermally induced changes falls within the range of natural variabilities of hydrologic properties under ambient conditions.

2.1.5 Dripping onto Waste Packages

The principal source of water contacting the waste packages is dripping from the drift ceiling or walls, either directly from the fractures or following condensation of water vapor onto the walls. This factor only reflects the volume of water falling on the waste package, not the location of the seeps, which is contained in the seepage into drifts factor (Section 2.1.3).

The specific analyses for this factor, as is shown in Section 5.12.4 of CRWMS M&O (1998a), show a low sensitivity in the TSPA. However, for the same reasons as discussed for the seepage factor, the uncertainty in this factor is dependent on the uncertainty in infiltration, percolation, and seepage, and should be viewed as having a moderate rating in determining system performance.

2.1.6 Humidity and Temperature at Waste Package

This factor is potentially important because water can also contact the waste packages by condensing from the water vapor in the emplacement drift and collecting under salts on the surface of the waste package, resulting in humid-air corrosion. Analysis of the sensitivity of waste package degradation due to different thermo-hydrologic environments was completed using temperature and relative humidity histories from various regions in the repository. The analysis showed that the variable temperature and relative humidity at the waste package surface did not significantly affect long-term waste package performance (Section 5.13.4, CRWMS M&O 1998a). Therefore, the importance of this principal factor to system performance is considered low.

2.2 LONG WASTE PACKAGE LIFETIME

2.2.1 Chemistry of Water on the Waste Package

A critical parameter for corrosion of the waste package materials is the chemistry of the water in contact with them (e.g., pH, chloride content). Performance assessment sensitivity studies indicate that this factor is highly important to the calculated peak dose rates. The waste package failure rate was significantly higher in the alkaline dripping case than in the base case (Section 5.12, CRWMS M&O 1998a). These studies are based on a very conservative representation of effects of pH on the waste package and may overestimate the importance of the chemistry. Considering a more realistic representation of these effects, the importance of the chemistry to postclosure system performance is moderate.

2.2.2 Integrity of Outer Waste-Package Barrier

Performance assessment sensitivity studies indicate a very high sensitivity to the containment provided by the waste packages. The outer carbon steel barrier plays only an indirect role in containment, the principal containment is provided by the corrosion-resistant inner barrier. However, the outer barrier does play a role. The outer barrier provides structural integrity to the package. Thus the effects of rockfall or other mechanical degradation modes are expected to be mitigated by this barrier. In addition, the outer layer provides additional protection from conditions that might accelerate the corrosion of the inner waste package layer. Information about waste package degradation is abstracted into models of total system performance as waste-package failure history, number of pit perforations per package, and average number of patch openings per package without explicit consideration of the performance of the outer barrier (Section 11.2.4, CRWMS M&O, 1998a). When consideration is given to the potential of this barrier to increase waste package lifetime, particularly in the first 10,000 years of the repository, the ultimate dose calculation is expected to be moderately sensitive to the lifetime of the outer carbon steel.

2.2.3 Integrity of Inner Waste-Package Barrier

The performance assessment sensitivity studies indicate a very high sensitivity to the containment provided by the waste packages. The judgement is that the performance up to one million years is highly sensitive to the lifetime of the corrosion-resistant inner barrier (Section 5.13.4, CRWMS M&O 1998a).

Some waste packages will contain barriers in addition to those identified here. For example, the high level waste packages will also include stainless steel canisters for the vitrified waste. The performance assessments did not explicitly evaluate these other barriers and they are not considered here; however, they are not precluded from future evaluations or credit that might be taken for them.

2.3 LOW RATE OF RELEASE OF RADIONUCLIDES FROM BREACHED WASTE PACKAGES

2.3.1 Seepage into the Waste Package

The fraction of water actually seeping into the breached waste packages determines how much of the waste form is released as a result of contact with flowing water. However, there is currently no experimental or observational information upon which to base the assumptions about how much water enters the waste package, and the assumed amounts are selected to be conservative bounds. Thus, a variation in these assumptions to include a less conservative range of seepage into the waste packages could significantly affect repository system performance. In addition, for similar reasons that the other characteristics associated with the flow system are moderately important to postclosure performance, a more realistic representation of seepage into the waste package could lead to it being moderately important to the performance of the system.

2.3.2 Integrity of Spent Fuel Cladding

Most of the spent fuel has Zircaloy cladding that is resistant to corrosion under reactor operating and pool storage conditions. Zircaloy is also likely to be an effective barrier to water that might enter the waste package for those cases where the cladding is intact. The performance assessment sensitivity studies indicate that this factor is very important to performance (Section 6.6.1, CRWMS M&O 1998a) for those waste packages containing competent, Zircaloy-clad commercial spent fuel. To evaluate the impact of variation in cladding on system performance, analyses were run with no cladding credit and with forced failure of the cladding by 10^5 and 10^6 years (Section 6.6, CRWMS M&O, 1998a). The results showed a significant impact on repository system performance, suggesting this factor is highly important to postclosure performance.

2.3.3 Dissolution of Spent Fuel and Glass Waste Forms

The mobilization of radionuclides contacted by water is constrained by the finite rate of alteration and dissolution of the solid waste form. Because technetium-99 and iodine-129 are soluble in the water at Yucca Mountain, their rates of dissolution are constrained only by the rate of degradation of the waste form. In addition, the possibility of alteration of the waste form to secondary phases that are not readily mobilized have not yet been included in the TSPA. This factor, in addition to the sensitivity analyses performed for this factor, indicates that it is of moderate importance to the analysis of repository performance (Section 6.6.1, CRWMS M&O 1998a).

2.3.4 Solubility of Neptunium

The mobilization rate of neptunium is primarily determined by its solubility. Because the contribution to the dose rate from neptunium-237 is calculated to become more important in later times, this factor is considered to be moderately important to postclosure performance of the system (Section 6.6.1, CRWMS M&O 1998a).

2.3.5 Formation and Transport of Radionuclide-Bearing Colloids

Radionuclide-bearing colloids are anticipated to form during degradation of the waste form and radionuclides also may become attached to naturally occurring colloids already in the water. Some studies show that colloid-facilitated transport may be efficient, and thus could provide a mechanism for enhancing the transport of radionuclides. The performance assessment sensitivity studies indicate that formation and stability of radionuclide-bearing colloids are moderately important to postclosure performance (Sections 7.2.5 and 8.5.2, CRWMS M&O 1998a).

2.3.6 Transport Through and Out of the Engineered Barrier System

Transport mechanisms in the waste package could affect the rate at which mobilized radionuclides would be released from breached waste packages. Further, if water in the waste package is limited, the radionuclides are likely to migrate only in films of water on the surface of the internal components of the waste package, and these films may not be continuous. Transport may be further inhibited by sorption onto corrosion products in the openings in the waste package through which the radionuclides must travel and on the outside of the waste package. Finally, the corrosion products and other materials from the engineered barrier system that fall to the bottom of the drift might further alter the transport time of the waste. These considerations have led the analysts to assign moderate importance to this principal factor.

2.4 RADIONUCLIDE CONCENTRATION REDUCTION DURING TRANSPORT AWAY FROM WASTE PACKAGES

2.4.1 Transport Through the Unsaturated Zone

Concentrations of radionuclides released from the waste packages could be reduced as the contaminated water from the waste packages mixes with uncontaminated water percolating through the unsaturated zone. Because the flow is predominantly in the fractures in the welded tuffs, the concentrations in that flow could also be reduced as radionuclides diffuse from the fractures into the pores of the rock matrix.

Sensitivity analyses were conducted to evaluate the significance of unsaturated zone radionuclide transport to the overall performance of the repository system (Section 7.6.1, CRWMS M&O 1998a). These analyses indicate that there is less than a factor of 5 change in the calculated dose rate from the "expected" value. These results indicate that reasonable ranges in the uncertainty in the key transport characteristics of the fractured tuffs (namely fracture porosity, matrix diffusion, retardation and colloid stability) have a low importance to dose rate. These results must be tempered, however, with some judgment associated with whether a significantly different conceptual model, such as one involving only fracture flow and transport, could or should be invoked.

The potential significance of an alternative conceptual model depends on many factors beyond the unsaturated zone, namely (1) dispersive effects in the saturated zone, (2) the location of any hypothetical well used as the point of withdrawal of the groundwater, and (3) the volumetric flow from the well. If one assumes that there is minimal dispersion in the saturated zone, that the well directly intersects the contaminated plume in the aquifer, and the well is sufficiently small that dilution of concentrations of the radionuclides by water away from the plume is very limited, then the calculated dose rate could be significant. Only if these conditions are met would the conceptual model of unsaturated zone transport be very significant to system performance, i.e., be given a high significance.

2.4.2 Flow and Transport in the Saturated Zone

Flow and transport in the saturated zone is a principal factor because it provides the way for dissolved and colloidal radionuclides to move away from the repository. The properties (e.g., Darcy flux, effective porosity, and pathways) determine how rapidly the radionuclides can travel away from the repository system. In addition, concentrations can be reduced through mixing of contaminated water from the repository with uncontaminated water in the saturated zone during transit and by mechanical dispersion (spreading of the contaminant plume). The performance assessment sensitivity studies indicate that this factor is moderately important to postclosure performance (Section 8.5.2, CRWMS M&O 1998a).

2.4.3 Dilution During Pumping

Additional dilution of radionuclide concentrations may occur at the wellhead if the pumping extracts a significant amount of water and mixes contaminated water with uncontaminated water. Considering possible ranges of pumping rates in the area, characteristics of the plume, and pumping interval of the well, this factor could be important to performance. This judgment is based on a comparison of a limited water user scenario with an analysis of mixing the potentially contaminated water in the alluvial aquifer by the large volume of water presently extracted in the Amargosa Farms area on an annual basis. This latter analysis assumes that the critical exposed population that will be using water from the alluvial aquifer several thousands and tens of thousands of years from now is similar to the current population.

It is likely that the applicable environmental standard will specify the location and characteristics of the populations and individuals to be addressed in the dose calculations. The significance of the potential dilution from pumping to system performance will be a function of the degree of dispersion in the saturated zone, the location of the compliance boundary, and the size and water consumption of the populations and groups of individuals considered. The judgement is that this factor is considered to have a moderate significance on postclosure performance given the assumptions that were the basis for the VA, and will need to be reconsidered as final regulations are promulgated for the Yucca Mountain site.

2.4.4 Biosphere Transport and Uptake

Models of uptake from the use of contaminated water for bathing or irrigation, uptake by plants and grazing animals, and aspects of the biosphere that could affect radionuclide concentration have been used in recent performance assessments (Section 9.7.2, CRWMS M&O 1998a). Analyses, using these models, indicate the potential for increase in or reduction in the radiological effect of the radionuclides. Biosphere pathways and their uncertainties need to be understood as part of creating a defensible description of overall system performance. The biosphere is not, however, considered a "barrier" as other key factors may be considered barriers. The performance assessment sensitivity studies indicate that this factor could be moderately important to the dose estimates (Section 9.7.4, CRWMS M&O 1998a). Given that the groups and individuals that will ultimately be considered in the consequence assessment have yet to be defined by the regulators, the biosphere transport factor in the analyses, although uncertain and potentially having an impact on performance, is currently considered to be of low significance.

3. POTENTIAL DESIGN OPTIONS

The principal factors discussed in Section 2 are for a specific design, but the design that will serve as the basis for the site recommendation and license application decisions has not yet been selected. An important activity is therefore the effort to select that design. Design options and alternatives that could improve performance and reduce the importance of uncertainties in the system are being evaluated in a comprehensive assessment to support initial design selection for site recommendation and license application.

As indicated in Section 2, the radionuclides of most importance to repository performance are those that are either potentially mobile or known to be mobile under expected repository conditions. Design options are important in selecting the system of natural and engineered barriers to contain and isolate these radionuclides. With respect to the potentially mobile radionuclides, design options will be considered to ensure that the system of natural and engineered barriers provides defense in depth in limiting their exposure to water, their mobilization, and their transport to the accessible environment. With respect to the mobile radionuclides, these options will play an even more important role. In this case defense in depth can only be provided by multiple engineered barriers. The design options therefore play a critical role in ensuring a system design adequate to address uncertainties related to release and transport of these radionuclides.

Three design options were considered as part of the VA analyses:

1. Backfill
2. Drip shields
3. Ceramic coatings on the waste package.

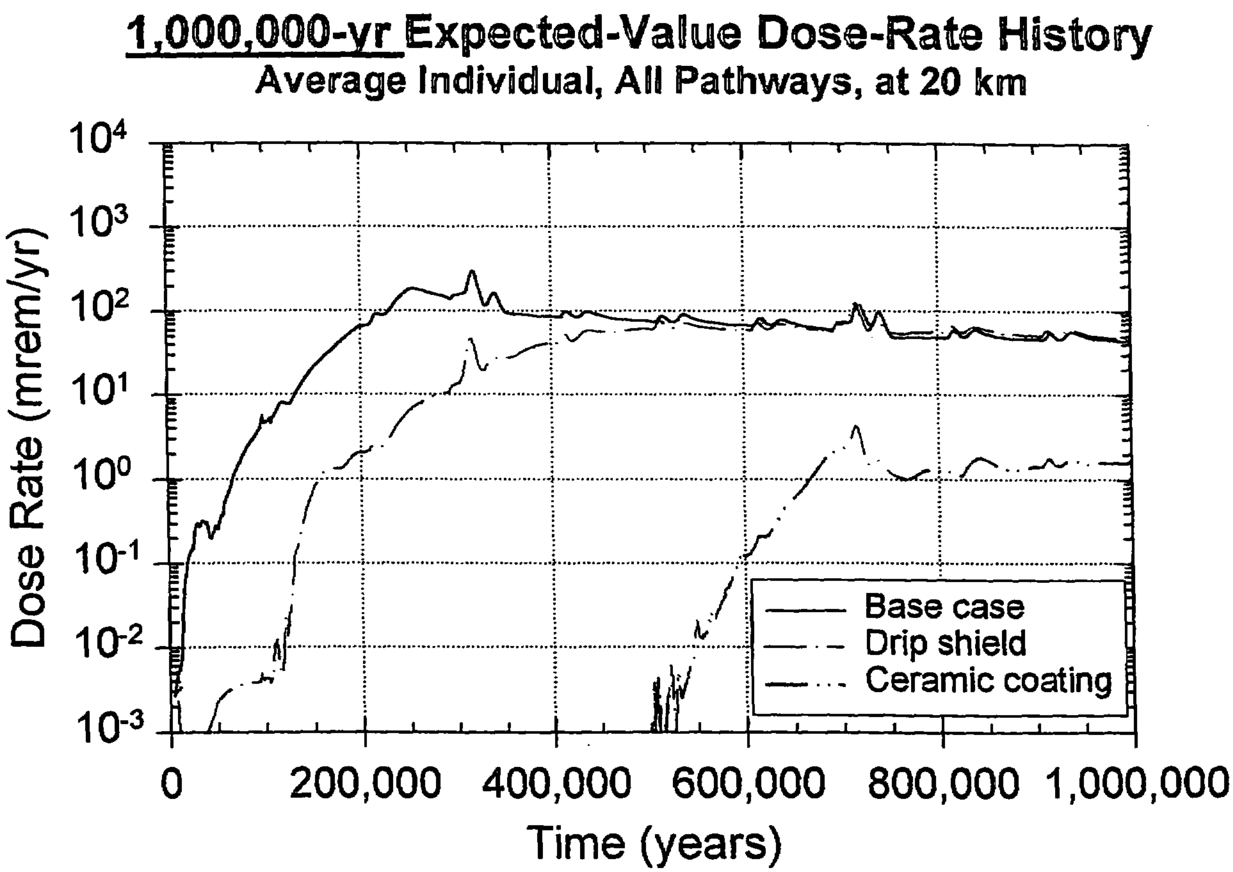
The results of analyses of performance for systems including these options are indicated in Figure 2. This figure shows the potential for greatly improved postclosure performance and for compensating for uncertainties in infiltration, percolation, seepage into drifts and transport of radionuclides to the biosphere at very long times (Section 5.13.4, CRWMS M&O, 1998a). Further, the analyses suggest that these options could be employed to provide defense in depth, i.e., redundancy in containing and isolating the waste.

The analyses of the design options are preliminary, and their feasibility and longevity have not yet been determined. The following sections summarize the potential importance of the VA design options, and the associated issues for which technical work is needed to support the initial design selection decision in 1999.

3.1 BACKFILL

Backfill could provide mechanical protection for the waste package and other engineered barriers from the effects of eventual drift collapse. Also, backfill would insulate the waste packages, resulting in higher temperatures and reduced relative humidity during the thermal period. In addition, backfill in concert with other barriers, such as drip shields, could prove an effective

Figure 2. Sensitivity of Postclosure Performance (Dose Rate) to Design Options



barrier decreasing the likelihood of water contacting waste packages. Sensitivity analyses did not show any significant differences in the overall waste package failure profiles between the no-backfill and backfill cases (Section 5.11, CRWMS M&O 1998a).

3.2 DRIP SHIELD AND BACKFILL

A drip shield placed above the waste package could divert seepage water away from the package as long as it remained intact. Backfill placed above and, perhaps, beneath a drip shield would protect it from mechanical damage. The backfill-drip shield combination could be highly effective at reducing the number of breached waste packages, and would therefore provide defense in depth.

There are several important uncertainties regarding the backfill-drip shield option. First, flow diversion around the waste packages depends on the longevity of the drip shield materials (e.g., ceramics and corrosion-resistant metal alloys) and stability of the backfill properties. The current performance assessment evaluated a single representative drip shield consisting of a 2-cm thick inverted U-shaped plate of Alloy 22 placed over the waste package. Information on the long-term stability of particular ceramics of interest has not been compiled, and bounding performance analyses have not been conducted. Second, flow diversion performance of the backfill-drip shield combination needs to be demonstrated for the full range of flow conditions, including low and high flow situations that could occur in the repository. Finally, the engineering feasibility of this barrier has yet to be demonstrated.

3.3 CERAMIC-COATED WASTE PACKAGE

Ceramic coatings on metals have been observed to have essentially no connected porosity, so that coatings on the waste package would be nearly impervious to water. It is likely, based on general chemical and physical properties of ceramics and observations of natural and man-made ceramics, that such coatings on the waste packages could remain intact for very long times at repository conditions (backfill may be required to mitigate the effects of rockfall). Such a barrier could have a significant effect on performance as long as it remains intact, and would protect the corrosion-resistant material under the coating. This approach would be highly effective at reducing the number of waste package failures over time and would provide defense in depth.

There are several significant uncertainties with this option. Stability against phase transitions and long-term continuity of the coating need to be evaluated. In addition, effects on the underlying metal need to be evaluated. If water is able to penetrate through the coating (e.g., at discontinuities) the chemical conditions at the underlying surface might become an issue. Although spray coating of ceramics on metals has been demonstrated for other purposes, the specifications for postclosure repository performance have not yet been determined. There are questions about the effectiveness of backfill in protecting such coatings against rockfall and other mechanical damage. Information in these areas will significantly increase confidence in the representation of ceramic coatings in performance assessment.

4. STATUS OF THE POSTCLOSURE SAFETY CASE

The purpose of the repository safety strategy is to provide the information to complete the postclosure safety case, the set of data and analyses regarding postclosure safety of the repository that will be used in DOE's decisions regarding site recommendation and license application. That case will focus on whether the repository system constitutes an unreasonable risk to the health and safety of the public, and will explain compliance with the regulatory objectives and criteria that will be specified for postclosure performance. As indicated in the current NRC regulations (10 CFR 60.101):

> "While these performance objectives and criteria are generally stated in unqualified terms, it is not expected that complete assurance that they will be met can be presented. A reasonable assurance, on the basis of the record before the Commission that the objectives and criteria will be met is the general standard that is required."

The elements of a competent postclosure safety case are those designed to provide reasonable assurance that public health and safety will be protected, and include the following:

- Assessment of expected postclosure performance and supporting evidence
- Design margin and defense in depth
- Consideration of disruptive processes and events
- Insights from natural and man-made analogs
- A performance confirmation plan.

The status of these elements of the postclosure safety case is summarized in the following sections.

4.1 ASSESSMENT OF EXPECTED POSTCLOSURE PERFORMANCE AND SUPPORTING EVIDENCE

4.1.1 Current Status

The current assessment of expected performance is being summarized in the Total System Performance Assessment-Viability Assessment: Technical Basis Document (CRWMS M&O 1998a). That assessment reveals that most of the radionuclides in the repository are not mobile at the Yucca Mountain site; they are insoluble or they sorb strongly to minerals and materials in the repository and cannot move out of the repository. A small fraction is relatively mobile, however, and could be transported away from the repository if contacted by water. This issue is mitigated for this small fraction if water is kept from contacting the waste in the first place.

Current data and analyses indicate that the Yucca Mountain site provides favorable features for limiting the contact of water with the waste. Its location in an arid region and the nature of the site itself limit the amount of water that can reach the repository. The site provides a thick unsaturated zone where the waste can be placed deep below the surface and yet well above the water table. Because of this, the waste packages would be protected from changes in conditions at the surface while being kept well away from groundwater. The site would, therefore, provide predictable and stable environments for design of engineered barriers that can further limit the exposure of waste to water.

Performance assessment studies show that, in addition to the natural barriers, engineered barriers could keep water away from the waste. They indicate, for example, that the highly corrosion-resistant inner container and the thick steel outer container of the reference design of the VA each will provide effective barriers against water for different conditions that will occur during the postclosure performance period. Although there are some issues that must still be addressed, current estimates indicate that robust waste packages could be designed to remain intact for hundreds of thousands of years in a range of expected repository environments. The base case modeling results for waste package degradation for the no-drip case show no waste package failure until about 700,000 years due to the resistance of Alloy 22 to humid air corrosion (Chapter 5,CRWMS M&O, 1998a). Under dripping conditions, the first waste package breach occurs at about 2,700 years, with only one percent of the packages failing by about 10,000 years, and about twenty-three percent of the waste packages failing by 100,000 years (Chapter 5, CRWMS M&0, 1998a). The studies also indicate that the spent fuel cladding would likely provide an additional barrier to water contacting the waste, even if both the outer and inner waste-package barriers were to be breached.

4.1.2 Nature of Work Needed to Complete the Case

Site characterization work has significantly reduced the number and magnitude of the uncertainties important to postclosure performance, but some issues still remain. These include uncertainties in the variability of the flow of water at the site and in the transport of radionuclides that might be released from breached waste packages. Some of these uncertainties can be reduced through additional technical work on the principal factors. Some of the issues can be addressed by invoking design measures to compensate for them as discussed in Section 3. The work needed to complete this case includes selection of the design for site recommendation and license application, identification of the principal factors for that design, and any technical work to address the remaining issues with these principal factors.

4.2 DESIGN MARGIN AND DEFENSE IN DEPTH

4.2.1 Current Status

The second approach to providing reasonable assurance of postclosure safety will be to demonstrate design margin and defense in depth[3]. In general, the repository system will rely on multiple engineered and natural barriers against the movement of water and radionuclides and

[3] Design margin refers to the inclusion of a margin of safety in the specifications for engineered components in order to account for uncertainty in the conditions and variability in the material properties. Defense in depth means the use of multiple barriers to mitigate uncertainties in conditions, processes, and events. In particular, the system of barriers is chosen so that failure in any one barrier does not result in failure of the system. An example of defense in depth is the utilization of a drip shield in addition to the robust waste package to keep water away from the waste: uncertainties in the performance of one of these features would be offset by performance of the other, increasing overall confidence in the system.

invoke other measures beyond those that can be explicitly demonstrated in a total system performance assessment. These multiple barriers and measures will be chosen to ensure that there is significant margin of safety and redundancy to address conditions that, while unanticipated, are sufficiently credible to warrant consideration. These include conditions that are modeled conservatively compared to the expected effects of the condition.

Explicit analyses to address design margin and defense in depth have not yet been conducted. Some of the performance assessment sensitivity studies, including those for the Viability Assessment, address this issue indirectly, in that importance of individual barriers has been explored in a preliminary fashion.

4.2.2 Nature of Work Needed to Complete the Case

The analysis of design margin and defense in depth of the repository system design will be conducted as a part of the evaluation of design alternatives and options and the initial selection of the design for site recommendation and the license application. This analysis will include both an analysis of processes or events that could affect system performance and an explicit evaluation of design margin and defense in depth with respect to the system performance measures. The first analysis will include an identification of the possible threats to performance and system features potentially capable of addressing those threats. That analysis will explicitly and transparently determine the effectiveness of the system features. The system evaluations of design margin and defense in depth will evaluate system performance for the designs under consideration, and assess both the uncertainty in that estimate and the degree of margin in meeting performance criteria. They will evaluate design margin in particular by considering the effect of neutralizing each of the barriers that contribute to performance. These analyses will transparently display the degree of reliance on individual elements of the system and the degree to which lower than expected performance of one barrier may be compensated for by the performance of another.

4.3 CONSIDERATION OF DISRUPTIVE PROCESSES AND EVENTS

4.3.1 Current Status

The postclosure safety case will also explicitly consider processes and events that could disrupt a repository at this site. These include disruptive natural processes (seismicity and volcanism), potential human intrusion associated with exploration for natural resources, and nuclear criticality.

Tectonics and Seismicity

Current information about the tectonic regime at Yucca Mountain suggests an upper bound to the mean ground acceleration at the site (Section 3.10, CRWMS M&O 1998b). Quantitative estimates of postclosure performance for this upper bound condition do not indicate any significant releases for these conditions (Chapter 10.5, CRWMS M&O 1998a). This work suggests that this issue can be addressed by current information.

Volcanism

Site characterization work has led experts to conclude that the likelihood of significant future volcanic activity near the site is negligible (Section 3.9, CRWMS M&O 1998b). Performance assessments that consider several extreme volcanism scenarios indicate no significant releases of radionuclides (Chapter 10.4, CRWMS M&O 1998a). This work suggests that this issue can also be addressed by current information.

Inadvertent Human Intrusion

Future human activity that might interfere with the repository cannot be precluded because human activity thousands of years into the future cannot be predicted. Performance assessment analyses indicate that drilling at the site could carry waste from a breached waste package down to the water table and, thereby, increase concentrations of radionuclides in the groundwater (Chapter 10.6, CRWMS M&O 1998a). However, mineral resource assessments indicate that the Yucca Mountain site does not exhibit characteristics that make it particularly attractive for exploration in the future (Section 3.11, CRWMS M&O 1998b). Therefore, although significant localized releases cannot be precluded for some scenarios, the information strongly suggests that the probability of these scenarios is very small. The National Research Council of the National Academy of Sciences (1995) suggested that human intrusion is a generic, not a site-specific issue, and that regulations for a Yucca Mountain Repository should require a single stylized calculation. Such a calculation would serve as an indicator of system robustness by evaluating the long term effects on the groundwater pathway from a single borehole that intersects a waste package and impacts the saturated zone. It appears that this issue can be addressed by current information.

Nuclear Criticality

All waste packages with criticality potential will include criticality-control measures (e.g., neutron absorbers). Therefore, criticality events will be prevented as long as the waste is maintained within the waste packages and the criticality measures remain in place (Chapter 10.6, CRWMS M&O 1998a). Nevertheless, analyses assuming a criticality event within a waste package were conducted (Section 10.6, CRWMS M&O, 1998a), and showed a potential change in the radioactive content of the waste package. The net effect on the radioactive content of the repository system was negligible (Section 10.8, CRWMS M&O, 1998a). Analyses have been conducted to address the possibility of a criticality event outside the waste package in the event of a breach of a waste package. These analyses indicate that the concentrations of dissolved radionuclides that are controlled by their solubility and the alteration rate of the waste form will be too low to result in the accumulation of a critical mass outside the waste package in less than a million years (Chapter 10.6, CRWMS M&O 1998a). The analyses have not yet addressed the potential for transport and accumulation of these radionuclides by way of colloids. Consequently, additional technical work is needed to address this possibility.

4.3.2 Nature of Work Needed to Complete the Case

Current information appears to be adequate to address most of the issues associated with the probability and consequences of disruptive processes and events that could affect postclosure performance. The additional technical work consists mainly of completing the development of the methods that will be used to evaluate criticality, and consolidating and documenting current information. Although the seismic regime will be considered further in developing site-specific seismic design requirements, it appears that current information is adequate to address postclosure performance issues (Chapter 10, CRWMS M&O 1998a). Likewise, current information is adequate for evaluating the postclosure effects of volcanism and inadvertent human intrusion for natural resource extraction, (Chapter 10, CRWMS M&O 1998a). Technical work regarding postclosure nuclear criticality is nearly complete, although work is needed to address the formation and stability of radionuclide-bearing colloids in the repository system and to finalize the methodology for addressing criticality in the regulatory arena.

4.4 INSIGHTS FROM NATURAL AND MAN-MADE ANALOGS

4.4.1 Current Status

Relevant information about the possible future waste isolation performance of the repository system can be gleaned from analysis of known natural or man-made systems that share characteristics with the repository system. Therefore, the fourth element of the postclosure safety case is to use knowledge gained from the study of natural and man-made analogs. A direct analog for a Yucca Mountain repository is not known; however, there are sites that can provide information on processes and conditions relevant to a repository system at the site. An important advantage of analogs is that processes can be studied that have been ongoing for very long time periods, and are potentially relevant to postclosure performance of the repository system.

The behavior of relevant materials and systems has been studied directly in a wide variety of natural and man-made settings. For example, analyses of the transport of dissolved species and the formation of minerals in modern or fossil hydrothermal systems have provided information on geologic and hydrologic processes operating for millions of years. Studies of natural radionuclide transport processes in uranium mining districts have been conducted (Section 6.3, CRWMS M&O 1998b). Man-made analogs, similarly, may provide information regarding relevant processes or materials over time scales and distances that are not reproducible in a laboratory or in limited-duration field studies. For example, examination of ancient ceramic human artifacts might provide useful information regarding the stability of candidate ceramic materials for drip shields or waste-package coatings. As with all site characteristics, natural analogs have significant limitations, including the incomplete and heterogeneous geologic record, uncertainty in characterizing the past conditions under which the processes took place, partial or imperfect analogy to repository conditions, and divergent interpretations of geologic data. However, as a supplement to site characterization and predictive modeling of repository performance, natural analogs offer the advantage of direct study of relevant processes over long time periods and extended spatial scales applicable to repository performance. Furthermore, data acquired from analog studies are generally independent of site characterization and modeling studies conducted for Yucca Mountain, and may provide a degree of independent verification of the reasonableness of selected aspects of the assessments of repository performance.

4.4.2 Nature of Work Needed to Complete the Case

Natural analog studies can provide information to help interpret the geologic and hydrologic conditions at the site, and the potential future evolution of the natural setting following construction of the repository. Analog studies may be most useful in analyses of the transport and deposition of radionuclides, the stability and transport characteristics of alteration minerals, the conditions in the shallow unsaturated zone, and the characteristics and behavior of man-made materials. All of this information could complement the short-term laboratory and field tests conducted at Yucca Mountain.

Of particular importance are studies of colloid transport in the field. The laboratory studies generally suggest that colloidal transport of radionuclides is not a significant issue, but information at field sites where radioactive contamination has been or is being studied suggests that the issue could be important. Work has been conducted at a number of sites that are being reviewed for relevance to Yucca Mountain, and there is full scientific cooperation between the DOE offices overseeing these field studies and the Yucca Mountain project.

To prepare for the site recommendation and license application decisions, a comprehensive review and summary of analog information potentially relevant to Yucca Mountain performance will be compiled. This review will include analogs for radionuclide solubility and geochemical processes that affect transport, that have been the focus of international projects supported by DOE in the past. The Peña Blanca site in northern Mexico will be evaluated as a natural analog to secondary precipitation of spent fuel radionuclides. Potential analog sites where precipitation and infiltration conditions are similar to Yucca Mountain, and where paleoclimatic information is available, will be included in the review. Geothermal areas, which may provide a means to exercise numerical models of coupled thermal-hydrologic-chemical processes, will be included. Colloidal transport of radionuclides at analog sites, both natural and anthropogenic, and man-made analogs of ceramic materials that may find application in the engineered barrier system, will be also be included. It is expected that the efforts to evaluate analogs in the field will be part of the performance confirmation program.

4.5 A PERFORMANCE CONFIRMATION PLAN

4.5.1 Current Status

The final element of the postclosure safety case is a performance confirmation plan for long-term testing and monitoring that would start before and continue after license application. Performance confirmation provides the means to address inherent limitations and uncertainties associated with performance and design analyses that will remain after site characterization. Performance confirmation will be designed to continue to monitor aspects of the performance of the repository after waste is disposed, for a period of at least 50 years after the initiation of disposal, and perhaps for much longer. Focused analog studies identified in the license application are to be carried out as part of performance confirmation. The information collected during this period could be more relevant for long-term analyses of the repository than any test results that will available for the license application. The increased understanding and confidence derived from long-term testing and observation will be of great benefit to decision makers, for example, to determine when to apply to the NRC for authorization to close the repository.

The performance confirmation plan that will be prepared for the license application will describe the tests to:

- Confirm that subsurface conditions encountered during construction, waste emplacement operations, and monitoring are within the ranges assumed in the license application
- Confirm that natural and engineered systems and components are functioning as intended and anticipated
- Confirm that modeling of processes is generally supported by analog studies
- Re-evaluate compliance with NRC postclosure performance requirements
- Evaluate the repository readiness for permanent closure.

A performance confirmation plan has been developed (CRWMS M&O 1997) that describes, in general terms, long-term testing to confirm the assessment of principal factors affecting postclosure performance. The plan is general because the specifics depend on the design features that will be selected and any corresponding changes to the postclosure safety case. The preliminary plan therefore provides only an outline of the areas to be addressed in testing during the period of construction, waste emplacement, and long-term monitoring, and needs to be augmented to include the more specific approach to using analogs described above (Section 4.4).

4.5.2 Nature of Work Needed to Complete the Case

Specific tests will be defined and the performance confirmation plan will be completed once the reference design for the license application has been chosen. That plan will define the activities necessary to address the elements of this design as specified in the requirements of 10 CFR Part 60, Subpart F or the requirements of a new NRC regulation when it is promulgated. The plan will specify monitoring, testing, analog, and analysis activities to be conducted to evaluate the accuracy and adequacy of the information used in the license application, in particular that information used to determine that the performance objectives will be met for the period after permanent repository closure.

The performance confirmation program defined in this plan will provide information on the coupled thermal, hydrologic, geomechanical, and geochemical processes that will occur in the repository system. Long-term thermal testing and observations of actual repository behavior that will be described in this plan will provide additional confidence beyond that presented in the license application based solely on site characterization up to that time.

The parameters and concepts identified for performance confirmation will be based on the understanding of natural and engineered barrier processes available at the time of submittal of the license application, the mathematical models formulated for these processes, the computer codes that have been developed to simulate these processes, and the parameters required for these computer codes. The plan will describe the process of integrating the additional information obtained after submittal of the license application into these concepts and parameter representations.

5. PROSPECTUS

The Repository Safety Strategy has evolved as site information has increased, repository system design has evolved, and performance assessment models have become more sophisticated. The original strategy in the Site Characterization Plan (DOE, 1988) addressed many existing issues in each of hundreds of factors potentially important to postclosure performance—the current strategy now focuses on a few issues in each of only nineteen principal factors. Design options and alternatives offer the potential to both directly improve performance and to provide still greater focus for the strategy.

The next most important steps in the project will be the comprehensive evaluation of design options and alternatives, selection of the design that will serve as the basis for the site recommendation and license application decisions, and associated allocation of performance to the components of that design and the natural system. These steps will result in the identification of key remaining issues that must be addressed before the site recommendation and license application decisions. Accordingly, when those steps are completed, this strategy will be reviewed to determine needs reflecting the selected design, the performance allocation, and the associated key uncertainties. That review could result in an important update to the Repository Safety Strategy. Any such update would necessarily continue to emphasize the need for a balanced approach to the resolution of critical uncertainties: testing and analyses to reduce uncertainties, design features to compensate for them, and performance confirmation to address issues that are most appropriately addressed in long-term testing. Equally important, it would provide the critical step in defining the overall logic for the postclosure safety case and defining the final steps necessary for completing that safety case.

6. REFERENCES

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APPENDIX A

OVERVIEW OF THE REFERENCE DESIGN FOR THE VIABILITY ASSESSMENT

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. The nineteen principal factors reflect the current understanding of the site and the most recent sensitivity studies, based on the VA reference design. The following text summarizes the key features of the VA reference design as they relate to the expected behavior of the repository system.

The reference design includes a waste package consisting of two barrier metals: an outer metal consisting of 10 cm (4 in.) of low carbon steel and an inner metal of 2 cm (0.8 in.) of corrosion-resistant high-nickel alloy ASTM B 575 N06022 (Alloy 22).

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
- U.S. Navy spent nuclear fuel and stabilized excess weapons-grade plutonium
- DOE-owned spent nuclear fuels dominated by the N-reactor fuels from Hanford
- 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 Idaho National Engineering and Environmental Laboratory, Idaho.

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

Additional features of the repository and engineered barrier system reference design that influence the long-term performance of the disposal system include the following:

- Areal thermal load, which corresponds to the spacing between waste packages and 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).