Viability Assessment of a Repository at Yucca Mountain

License Application Plan and Costs

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Viability Assessment of a Repository at Yucca Mountain

Volume 4: License Application Plan and Costs

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ACRONYMS

ASTM	American Society for Testing and Materials
BLM	U.S. Department of the Interior, Bureau of Land Management
CFR	*Code of Federal Regulations*
CRWMS	Civilian Radioactive Waste Management System
DOE	U.S. Department of Energy
EIS	Environmental Impact Statement
EPA	U.S. Environmental Protection Agency
LA	License Application
M&O	Management and Operating Contractor
NRC	Nuclear Regulatory Commission
OCRWM	Office of Civilian Radioactive Waste Management
TSPA	Total System Performance Assessment
USGS	U.S. Geological Survey
VA	Viability Assessment
YMP	Yucca Mountain Site Characterization Project
YMSCO	Yucca Mountain Site Characterization Office

Measurements

Btu	British thermal unit
cm	centimeter
Eh	redox potential
ft	foot
g	gram
in.	inch
kg	kilogram
km	kilometer
kPa	kilopascal
kV	kilovolt
kVA	kilovolt-ampere
lin ft	linear feet

m	meter
mm	millimeter
MPa	megapascal
MTHM	metric tons of heavy metal
MTU	metric tons of uranium
MVA	megavolt-ampere
nm	nanometer
pH	hydrogen-ion concentration notation
ppm	parts per million
ppmv	parts per million by volume
psi	pounds per square inch
rem	roentgen equivalent man
wt	weight

OVERVIEW

Volume 4 provides the U.S. Department of Energy (DOE) plan and cost estimate for the remaining work that must be accomplished to proceed from the Viability Assessment (VA) to submittal of a license application (LA) to the Nuclear Regulatory Commission (NRC). This work includes an evaluation of the suitability of the Yucca Mountain site for development as a geologic repository and preparation of an environmental impact statement (EIS). Both steps are components of the documentation required to support a decision in 2001 by the Secretary of Energy on whether to recommend to the President that the site be approved for development as a Monitored Geologic Repository. If the President recommends the site to the U.S. Congress and the site designation becomes effective, DOE will submit to NRC an LA for authorization to construct a repository.

The remaining work described in Volume 4 constitutes the last iteration of site characterization, repository design, and evaluation of the probable performance of a Monitored Geologic Repository in the Yucca Mountain geologic setting (see Volume 1, Section 1.7.3). The plans for the next 4 years of work as presented in this volume are based on the results of the previous 15 years of work, as presented in Volumes 1, 2, and 3. Volume 1 describes what DOE has learned to date about the Yucca Mountain site. Volume 2 describes the current reference repository design, several design options that might enhance the performance of the reference design, and several alternative designs that represent substantial departures from the reference design. Volume 2 also summarizes the results of tests of candidate materials for waste packages and for support of the tunnels into which waste would be emplaced. Volume 3 provides the results of the latest performance assessments undertaken to evaluate the performance of the design in the Yucca Mountain geologic setting. DOE used the results described in Volumes 1, 2, and 3 to identify and prioritize the remaining technical work needed to ensure that a repository at Yucca Mountain would protect public health and safety and the environment during the preclosure period of operations and after the repository has been permanently closed and sealed. DOE believes that information gained from conducting the planned work, together with information already obtained, will support a site suitability evaluation for the site recommendation and, if the site is recommended and designated, a defensible LA.

Volume 4 is divided into seven sections. Section 1 is a brief introduction. Sections 2 and 3 present the rationale for the remaining technical work and plans for the remaining technical work, respectively. Section 4 addresses the statutory and regulatory framework within which the work must be done and the decisions that must be made. Sections 5, 6, and 7 deal with support activities, costs, and schedules, respectively. Appendix A is a glossary of technical terms used in this volume.

If funding is available to cover the estimated costs in Section 6, DOE will proceed with the work described in Section 3 in accordance with the schedule in Section 7 and will submit an LA to NRC in 2002.

RATIONALE FOR TECHNICAL WORK NEEDED TO COMPLETE THE LICENSE APPLICATION

DOE investigations of the Yucca Mountain site have resulted in a substantial understanding of the site, a preliminary reference repository design, and assessments of the performance of the repository system. These work products are of the type needed to support site recommendation and submittal of an LA. However, additional technical work is needed to complete the postclosure safety case, support the preclosure safety case, and support remaining design decisions.

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. DOE guidelines for the evaluation of site suitability and NRC disposal regulations both recognize that there are uncertainties inherent in evaluating the performance of a repository thousands of years into the future. DOE must demonstrate, with reasonable assurance, that the repository will satisfy the regulatory objectives and criteria for postclosure performance to obtain a license from NRC to construct and operate a repository. The postclosure safety case comprises multiple lines of evidence that, collectively, are intended to provide the requisite reasonable assurance. As planned, the postclosure safety case will include the following five elements:

- 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

Assessment of expected postclosure performance and supporting evidence. The current assessment of expected performance (summarized in Volume 3) indicates that the vast majority of the radionuclides in the waste that would be disposed of are not mobile at the Yucca Mountain site. These radionuclides are either insoluble or would sorb strongly to minerals and materials in the repository and would, therefore, be unlikely to migrate away from the repository. However, a small fraction of the radionuclides in the waste is relatively mobile and could be transported away from the repository by water.

Current data and analyses indicate that the Yucca Mountain site provides favorable features for limiting the contact of water with the waste. The site is located in a semiarid region, with precipitation averaging about 7 in./year. Of this precipitation, only about 5 percent seeps into the ground, because the remaining 95 percent either runs off, evaporates, or is taken up by desert vegetation. Future climates might be wetter than the climate today, and the precipitation might double. Still, only a small fraction of the precipitation would seep into the ground. The Yucca Mountain site provides a thick unsaturated zone of rock where the waste can be placed deep below the ground surface, yet well above the local water table. The unsaturated zone provides a predictable environment for which engineered barriers can be designed to further limit the exposure of waste to water.

Performance assessment studies show that engineered barriers could keep water away from the waste for thousands of years. The studies indicate, for example, that the highly corrosion-resistant inner layer and the thick steel outer layer of the reference design waste package could provide effective barriers against water under different conditions that could occur during the postclosure performance period (see Volume 2, Section 5.1.2). Although some issues must still be addressed, the analyses indicate that robust waste packages could be designed to remain intact in the repository environment for thousands of years. The studies also indicate that the metal cladding on most spent nuclear fuel would likely provide an additional barrier against exposure of waste to water.

Design margin and defense in depth. Because there will always be uncertainties in estimates of repository performance over many thousands of years, DOE is not relying solely on such estimates to demonstrate reasonable assurance that the postclosure performance standards will be met. Design margin and defense in depth provide important additional assurance.

"Design margin" refers to the standard engineering practice of including capacity in excess of design requirements to provide margins of safety in specifications for engineered components. Design margin is required to account for uncertainty in the conditions to which the components will be subjected and for variability in the properties of component materials. "Defense in depth" is the term used to describe the property of a system of multiple barriers that are diverse, independent, and redundant such that failure by any single barrier will not result in failure of the entire system. The outer and inner layers of the waste package would have different corrosion modes and are an example of defense in depth. Other barriers, such as drip shields, backfill, and ceramic coatings to the waste package could be added to the reference design to provide additional design margin and defense in depth.

Site characterization and laboratory testing have resulted in a reasonable understanding of the environments in which the engineered barriers would perform. Based on this understanding, DOE will analyze the design margin and defense in depth provided by the reference design, design options, and alternative designs to support the selection of the repository design that will be carried forward for site recommendation and submittal of the LA.

Consideration of disruptive processes and events. Compliance with the postclosure dose standard cannot be reasonably assured without explicitly considering those processes and events that could conceivably disrupt the repository and compromise its expected performance. DOE has identified four such potentially disruptive processes and events at the Yucca Mountain site: earthquakes, volcanoes, inadvertent human intrusion during exploration for natural resources, and nuclear criticality.

Yucca Mountain is located in the southern Great Basin, a region that is subject to earthquake activity. Investigations of seismic hazard at the site indicate that potential earthquake ground shaking can be accommodated by standard nuclear design and construction practices and that repository facilities can be located to avoid any faults that have a potential for significant movement. Performance assessments conducted to date indicate that earthquake activity would have little or no impact on long-term repository performance.

The large-volume, silica-rich volcanism that created the rock of Yucca Mountain ceased in the region about 7.5 million years ago. This type of volcanism was succeeded by relatively low volume basaltic volcanism, as evidenced by volcanic cinder cones near Yucca Mountain. The geologic record indicates that the rate of basaltic volcanic activity is declining, with the last episode occurring about 75,000 years ago. Statistical analyses of the pattern and rate of cinder cone activity around Yucca Mountain over the last million years indicate that the average risk of a volcanic disruption at the site in any given year is approximately 1 in 70 million. In addition, performance assessments indicate that most volcanic disruption scenarios would have little impact on postclosure repository performance.

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 peak dose rates would increase in a scenario in which exploratory drilling penetrates a waste package and waste is carried down the drillhole to the water table. However, mineral resource assessments indicate that the Yucca Mountain site does not exhibit characteristics that would make it attractive for exploration.

A nuclear criticality is a non-explosive nuclear chain reaction that occurs when enough fissionable material comes together in a precise manner to form a "critical mass" and the required conditions exist to allow the chain reaction to proceed. Waste packages will contain neutron-absorbing materials to preclude criticality events for as long as the waste is maintained within the waste packages. Performance assessments indicate little chance that leaching of waste by downward percolating water could construct a critical mass of fissionable material after the waste packages have been breached. However, the potential for transport and accumulation of radionuclides by way of colloids has not yet been completely analyzed, and this work remains to be done. Performance assessments indicate that a nuclear criticality, if it were to occur, would have little impact on postclosure performance.

Insights from natural and man-made analogs. A direct analog for a repository at Yucca Mountain does not exist; however, there are sites that can provide information on processes and conditions relevant to a repository system. Studying such natural analogs may provide information that bears on the transport of radionuclides over extended periods and distances that cannot be easily duplicated in laboratory or field experiments. Natural analogs can also provide information on the effect of various environmental conditions on materials that are intended to have a long life.

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 periods and extended distances applicable to repository performance.

Man-made analogs 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, ancient ceramic artifacts might provide useful information about the stability of candidate ceramic materials for drip shields or waste package coatings.

A performance confirmation plan. The final element of the postclosure safety case is a performance confirmation plan, the plan for long-term testing and monitoring. Performance confirmation begins with the documentation of current conditions, which is performed during site characterization, and continues until the repository is permanently closed. The DOE performance confirmation program is documented in the *Performance Confirmation Plan* (CRWMS M&O 1997d). This plan will be updated as needed and is intended to contribute to reasonable assurance when the LA is submitted by defining in advance the tests that will be conducted during construction and operation of the repository. The plan has the following objectives:

- Confirm that subsurface conditions encountered during construction, waste emplacement operations, and monitoring are within the ranges assumed in the LA

- Confirm that natural and engineered systems and components are functioning as intended and anticipated

- Evaluate compliance with NRC postclosure performance requirements

- Evaluate readiness for permanent closure

Some long-term tests, such as the drift-scale thermal tests and materials tests, are either in progress or will be started before the LA is submitted. These long-term tests are expected to continue after the LA has been submitted and to become parts of the performance confirmation program. Information obtained from these tests may be used to enhance repository design features, as appropriate.

Repository safety strategy. The DOE *Repository Safety Strategy: U.S. Department of Energy's Strategy to Protect Public Health and Safety After Closure of a Yucca Mountain Repository* (DOE 1998b) describes how a repository at Yucca Mountain is expected to contain radioactive wastes for thousands of years and ensure that radiation doses to persons living near Yucca Mountain would not exceed regulatory limits. By providing a conceptual model of how the natural and engineered barrier systems would work together to meet the postclosure performance objectives, the repository safety strategy provides a framework for integrating site information, repository design information, and performance assessment results. This, thereby, provides a framework for identifying the information needed to complete the postclosure safety case.

The repository safety strategy (DOE 1998b) identifies four key attributes of an unsaturated repository system that are important to meeting postclosure performance objectives:

- 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

These 4 key attributes are associated with 19 principal factors of greatest importance to postclosure performance (see Table 2-1). To identify information needs and establish work priorities, the repository safety strategy considers the relative importance of each principal factor to postclosure repository performance, what is known now about each principal factor, what more can be learned, and what knowledge is desirable by the time of site recommendation and submittal of the LA. Because the 19 principal factors reflect, in part, the VA reference design, and because DOE is still evaluating design options and alternative designs, the repository safety strategy also includes the identification and performance of work needed to select the repository design for the site recommendation and LA and reevaluation of the principal factors.

The principal factors described in this volume are a focus of much of the remaining technical work needed to complete the postclosure safety case. Section 2.2.3 describes the principal factors and provides an evaluation of the potential significance of each factor to postclosure repository performance. Section 2.2.4 presents priorities for remaining technical work, based on what is known now about each principal factor, what more can be learned, and what knowledge is desirable to support site recommendation and the LA. Section 2.2.5 discusses design options for the VA reference design that have been examined in performance assessment sensitivity studies. These options represent ways that the performance of the reference repository system might be improved. Section 2.2.6 summarizes the specific technical work planned to address the prioritized principal factors and to support design selection.

Preclosure Safety Case

The second major objective for planned technical work is development of information required to support the preclosure safety case. The preclosure safety case comprises the body of evidence necessary to demonstrate that worker and public health and safety will be protected while waste is being emplaced and monitored, before the repository is permanently closed. The key elements are systematic evaluation of design basis events; classification of structures, systems, and components that are important to safety; verification of system design for compliance to requirements; and the use of demonstrated technology and accepted design criteria.

Systematic identification of design basis events. There are two design basis event categories defined in the governing NRC regulation, 10 CFR 60. Category 1 describes "those natural and human induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area." Category 2 consists of "other natural and human induced events that are considered unlikely, but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety."

DOE is categorizing design basis events by frequency screening of sequences of events. This process starts with a hazards assessment that identifies initiating events. Event-tree modeling is applied to define potential accident sequences (or scenarios) that could result in release of radionuclides to the environment. This event-tree modeling describes an accident sequence as an initiating event followed by one or more events that propagate, or fail to mitigate, the event sequence so that radionuclides are released. The frequency for such sequences is the product of the frequency of the initiating event times the conditional probability or probabilities of occurrence of each subsequent event in the sequence.

Worker and public preclosure radiation dose limits are defined in 10 CFR 60 for Category 1 and Category 2 design basis events. Satisfying these dose limits is the driving constraint for developing the preclosure radiological safety case.

Classification of structures, systems, and components important to radiological safety. DOE has completed a classification analysis for the repository system that determined what structures, systems, and components in the repository affect quality, based on their radiological safety-related functions. This classification analysis will be augmented by identifying and analyzing the design basis events to support a graded quality assurance classification of structures, systems, and components important to radiological safety before the site recommendation and the LA are submitted.

Verification of system design for compliance to requirements. Safety analyses will ensure compliance with 10 CFR 60 requirements. The safety functional requirements of the structures, systems, and components important to radiological safety are included in applicable design requirements documents. Quality assurance procedures require design reviews to ensure the design requirements have been correctly selected and incorporated in the design. Qualification testing and/or evaluation will be performed in accordance with quality assurance procedures to verify that the design is in compliance with the design requirements.

NRC is currently revising regulations applicable to a repository at Yucca Mountain. If revisions to the regulations indicate that more safety analyses are necessary, the work will be performed and factored into the system description documents as appropriate. The design will then be verified and updated as necessary to ensure compliance with NRC regulations.

Use of demonstrated technology and accepted design criteria. The Monitored Geologic Repository will be the first facility licensed under 10 CFR 60, but certain aspects of nuclear facilities already licensed under 10 CFR 50 and 10 CFR 72 may be applicable.

DOE has established a formal compliance program to assess applicability and ensure compliance with NRC regulatory guidance documents and related industry codes and standards. This assessment, along with supporting rationale, is formally documented as input to design criteria via compliance program guidance packages. The compliance program provides guidance for planning future activities, for developing engineering design guides, and for analyzing designs related to design basis events. The compliance program also supports developing acceptance criteria for work in non-engineering topical areas considered in developing design criteria for engineered systems.

Information needs for site recommendation and licensing. Identification and evaluation of design basis events and their radiological consequences during preclosure operations are needed for site recommendation and the LA. Radiological safety analyses (i.e., hazard analyses, design basis events analyses, radiological consequence analyses, and nuclear criticality analyses) comprise an ongoing effort to determine these consequences. Analyses that will be completed before submittal of the site recommendation and the LA include the assessment of credible internal (e.g., drops) and external (e.g., earthquake and rainfall) events resulting in a final list of design basis events that the repository must be able to prevent or mitigate. Specific remaining information needed to enable completion of the radiological safety analyses is listed in Section 2.3.3.

Additional Technical Work Needed to Support Design Decisions

A major thrust of the technical work remaining is to conduct evaluations, including comparisons of options and alternatives, leading to selection of the design to support the site recommendation and the LA. Selecting the design involves a sequence of decisions regarding criticality issues, approaches to repository sealing and closure, and evaluation of design alternatives.

Criticality issues. Current regulations require that nuclear criticality be prevented or its potential minimized in all phases of repository operation and after permanent closure. Two important aspects of the criticality analysis are securing NRC approval for using burnup credit to demonstrate criticality control and evaluating the effectiveness of measures to control criticality. Work is planned to support both of these aspects.

Approaches to repository sealing and closure. The technical and environmental aspects of permanently closing the repository would be subject to NRC approval of a DOE application to amend its license to allow for permanent closure and decommissioning of the repository. However, the general approach will be evaluated before the LA is submitted to ensure that designs, construction, waste emplacement, and other operations do not preclude permanently closing the repository. Technical work will include evaluating alternative approaches and closure activities including the possibility of a longer monitoring period and possible attendant maintenance needs. The work will address sealing all underground openings, decommissioning surface facilities, reclaiming the site, establishing institutional controls, and planning for postclosure performance monitoring.

Technical work to evaluate design alternatives. A comprehensive and systematic design process will be conducted leading to selection of the design that will be the basis for the site recommendation and LA decisions. The design process will consider the potential advantages of design options and alternative design features and design concepts.

The reference design was developed to provide a consistent basis for making and comparing performance assessment evaluations. Along with the reference design, DOE identified and is evaluating three design options with the potential to enhance performance. These three options are backfill, drip shields, and ceramic coating.

In preparation for the licensing process, DOE is factoring additional considerations into the design selection process:

- Are there fundamentally different (alternative) repository design concepts that could meet performance standards more effectively than the reference design?

- Are there design features that could be added or incorporated into either the reference design or any alternative concept(s) with significant benefit?

- Are there alternative concepts or features that, in addition to meeting performance standards, could provide additional advantages with regard to operational and regulatory and issues?

Fiscal year 1998 work was directed at planning and beginning the comprehensive evaluation to select an initial LA design by May 1999. Updated and improved site models and data, new performance assessment evaluations, new engineering analyses, and new cost estimates will be prepared as needed to provide answers to the above questions.

TECHNICAL WORK PLANS

Volume 4 summarizes the DOE technical work plans for the time remaining before a decision is made on site recommendation and the LA is submitted. This period represents a transition from primary focus on scientific investigations to primary focus on design, engineering, and performance assessment. The work is organized under the major headings of site investigations, design, and performance assessment.

Site Investigations

The purpose of the remaining investigations is to reduce uncertainties in site features, processes, and events important to repository performance. The work scope addresses two broad areas: expected conditions, and disruptive processes and events, such as earthquakes, that could change the natural system and the way water flows through it. The work categories include the following:

- Geologic framework and disruptive events

- Unsaturated zone processes

- Saturated zone processes

- Thermal testing

- Near-field environment and coupled processes

- Performance confirmation

Geologic framework and disruptive events. This work includes improving the characterization of the geologic framework of the site and improving the evaluation of natural processes and events that could change the physical and chemical properties of rock and water at the site. Much of the effort concentrates on incorporating new data into and refining the three-dimensional model that represents the rock formations and geologic structures comprising Yucca Mountain. This work includes studies providing additional site-specific soil and rock property data needed to finalize repository design to account for potential seismic disturbances.

Unsaturated Zone Processes. This work addresses changes in climate and their effects, infiltration of water through surface soils and near-surface rocks to rock layers below, and movement of water downward through the unsaturated zone (the rock mass above the water table). The objectives are to continue to monitor and improve the understanding of how and at what rates water passes through the unsaturated rock and to improve predictions of the conditions under which water may seep into the waste emplacement drifts.

In addition, this work addresses factors that affect transport of any radionuclides released from a repository through the unsaturated zone. The planned work focuses on the processes of advective flow, matrix diffusion, and sorption. Advective flow involves flow through fractures. Matrix diffusion involves the movement of radionuclides from fractures in the rock through the internal structure, or matrix, of the rock. This redistribution of radionuclides slows their movement, because water flows more quickly through the fractures than through the rock matrix. In sorption, radionuclides are attracted to, and held for a time by, other particles or minerals present in the rock. The effects of sorption processes are positive when they retard the transport of radionuclides but can also be negative if radionuclides that otherwise would not be transported sorb onto colloids (tiny particles suspended in the groundwater) and the groundwater carries the colloids to locations where people can be exposed. The objective of this work is to improve the understanding of these processes that can reduce the concentration of radionuclides in unsaturated zone water.

Remaining work will focus on evaluating the transport implications of radionuclide-bearing colloids, completing a field test at Busted Butte to examine transport through rock similar to that underlying Yucca Mountain, and conducting investigations within the Exploratory Studies Facility and the cross drift to better understand how water is moving through the mountain and behaving near excavated openings.

Saturated Zone Processes. Groundwater flow provides a primary path for radionuclide transport once the radionuclides reach the saturated zone. This work addresses the flow of water in the saturated zone, located below the water table. The results will provide additional information about the features of the rock that allow, enhance, or retard the movement of water. This information will be factored into models that represent the rock layers and the water flowing through them. The way water mixes in the different layers in the saturated zone is of particular interest because it controls the amount by which radioactive elements (radionuclides) in the groundwater will be diluted, which in turn directly affects potential doses to exposed persons.

Remaining work will reduce uncertainties concerned with the hydraulic characteristics of rock units in the saturated zone, provide additional information on the chemistry of saturated zone waters, and implement a drilling program to create monitoring wells south of Yucca Mountain along the expected path that water will follow from beneath Yucca Mountain toward the Amargosa Valley. Additional drilling to characterize saturated zone hydrologic conditions may be planned if needed to support refinement of the saturated zone flow and transport model or to support performance assessment.

Thermal Testing. This work studies how the heat produced by deposited waste would affect mechanical, hydrological, and chemical characteristics of the emplacement drifts and surrounding rock. The objectives are to understand how mechanical, hydrological, and chemical processes are affected by heat and how they interact with one another under the influence of heat. The results will be factored into models that represent the rock layers and the heat flowing through them and provide additional information about the features of the rock that allow, enhance, or retard the movement of heat. The following three field tests are currently underway:

- Characterizing the large block of rock at the surface near Yucca Mountain that was heated for more than a year and is now in a cooling phase

- Completing the analysis of data from the single heater test, which heated rock within the Exploratory Studies Facility and then allowed the rock to cool

- Conducting a drift scale test that involves heating and monitoring a drift and surrounding rock within the Exploratory Studies Facility, and planning and implementing a field thermal test in the cross drift excavation

Near-Field Environment and Coupled Processes. This work studies how the natural environment within the immediate vicinity, or near-field environment, of the waste packages will be affected by excavation of the waste emplacement drifts and other underground openings, by the heat generated by the waste, or by the introduction of non-native materials. The work also will address the extent to which heat generated by the waste will create an altered zone of rock away from the waste packages. The studies will address the effects of coupled (or interacting) thermal, mechanical, hydrological, and chemical processes. In addition, studies will evaluate the growth of microbes under the thermal and chemical conditions within the emplacement drifts and the potential effect these microbes will have on waste package corrosion and radionuclide transport. The overall objective of this work is to better understand the conditions that will prevail in the near field, how they will change with time, and how they will affect the performance of waste packages and other engineered and natural barriers to radionuclide transport.

Remaining work includes laboratory tests and modeling to improve simulations of the performance of engineered barrier system design options; compare the performance of different materials that might be used in a repository; improve evaluations of the mechanical stability of ground support materials; refine evaluations of thermohydromechanical coupled effects on engineered barrier system design options and rock stability; refine evaluations of thermohydrochemical coupled effects on seepage into drifts and engineered barrier system performance; improve specification of the near-field chemical environment; and update models of flow and transport through the altered zone.

Performance Confirmation. This work consists of data collection and analysis tasks to test, confirm, and provide added confidence that models, bounds, and uncertainties used in design and performance assessment are correct. Performance confirmation activities include monitoring water level and earthquake activity and assessing the performance of seals.

Design

Those aspects of the design related to radiological safety and waste isolation must be completed to a level of detail that enables NRC to determine that the repository, as designed, will adequately protect the public and worker health and safety. Design options are being considered to enhance the performance of the present reference design. Design alternatives are also being studied that are a substantially different approach than the present reference design. The design effort is prioritized using a process called binning, in which repository systems are categorized in terms of their importance to radiological safety and waste isolation, and licensing precedent, into one of three categories as follows (in descending order of importance):

- Bin-3 items have implications for radiological safety or waste isolation and have no licensing precedent.

- Bin-2 items have implications for radiological safety or waste isolation and have licensing precedent.

- Bin-1 items have no implication for radiological safety or waste isolation.

The binning concept is described in detail in Volume 2, Section 2.3.

The three major areas of design work are subsurface, waste package, and surface design.

Subsurface design. Subsurface design encompasses the layout of the underground facility; ground control systems, such as concrete drift linings that ensure the mechanical stability of underground openings; ventilation; subsurface waste package handling; monitoring and control systems; waste retrieval systems; planning for the performance confirmation program; planning for sealing and permanently closing the subsurface facility; subsurface utilities; systems to remotely emplace and retrieve waste; and systems to maintain subsurface radiological safety. A key, near-term task is choosing the subsurface design options and alternatives to be incorporated into the LA design.

Waste package design. This work scope covers designing the containers that will hold the radioactive waste within the underground repository. All the structures, systems, and components associated with the waste packages are designated Bin 3, highest importance, because of their radiological safety and waste isolation implications, and lack of significant regulatory precedent. Work includes designing waste packages for the types of radioactive waste that would arrive at the repository in different forms and conditions. This includes additional development of processes for waste package fabrication and closure welding, further analyses to ensure waste packages will preclude nuclear criticality events, and further testing and modeling of the waste package and waste material, referred to as the waste form (see Volume 2, Section 5.1).

Surface facility design. The surface area of the repository consists of the radiologically controlled area, where radioactive material will be received and handled, and the remaining (balance-of-plant) area. The work scope includes design of the waste handling facility and surface waste handling systems; the Waste Treatment Building and systems to treat radioactive waste that is generated onsite; the Carrier Preparation Building, in which shipping casks would be prepared for removal from incoming trucks or railroad cars; and sitewide systems that serve the entire repository site, such as utility systems, safety and security systems, surface environmental monitoring, management and administration, maintenance, supply, and general site transportation. Surface facility design and operations are discussed in Sections 4.1 and 6.2 of Volume 2, respectively.

Performance Assessment

Much of the performance assessment work between now and the submittal of the LA will be directed at reviewing the improved process models resulting from current site investigation and design activities, and then abstracting or simplifying the component process models for use in the total system performance assessment (TSPA) for the LA. Process models are mathematical representations of features, events, and processes that could affect how the repository functions. Taken together, the abstractions of the process models comprise the TSPA model used to evaluate future doses of radiation to persons living near Yucca Mountain. There are process models for the interaction of heat and water in the repository; the near-field geochemical environment; waste package degradation; waste form degradation and the mobilization of radionuclides and other contaminants of concern by water; transport of radionuclides through engineered barriers; unsaturated zone water flow; transport of radionuclides through the unsaturated zone; saturated zone flow and transport; disruptive processes and events; and the movement of radionuclides throughout the biosphere, the region in which flora and fauna are present. The model abstractions will be updated to incorporate new information and refined process models about the site and updated repository and waste package designs. Performance assessment models and results are discussed in Volume 3.

The TSPA model will be used to estimate potential radiation doses to the potential population thousands of years in the future. These forecasts will support both the site recommendation and the LA. However, an equally important task will be to characterize the uncertainty in dose estimates and to evaluate the sensitivity of dose estimates to assumptions about how the repository is designed, or the natural system functions. These uncertainty characterizations and sensitivity studies will be key factors in selecting design options to enhance the current design or an alternative repository design. The design will be selected to achieve the design margin and defense in depth necessary to demonstrate reasonable assurance that the postclosure performance standard will be satisfied.

STATUTORY AND REGULATORY ACTIVITIES

In addition to the technical activities required to support the testing, design, and performance assessment work, a substantial body of other work is needed to comply with statutory and regulatory requirements. This work is related to environmental compliance and includes preparing an EIS, issuing a site recommendation, and conducting activities required by the NRC licensing process.

Environmental Impact Statement and Environmental Compliance

If the Yucca Mountain site is found suitable, the Secretary of Energy may recommend that the President approve the site for development of a repository. An EIS must accompany such a recommendation. The Nuclear Waste Policy Act of 1982 and other applicable laws also require DOE to comply with environmental regulations to protect the public health and the natural surroundings of Yucca Mountain.

Environmental Impact Statement. The EIS will follow criteria established by the Nuclear Waste Policy Act of 1982 and the National Environmental Policy Act of 1969, and rules (or implementing regulations) issued by the Council on Environmental Quality and DOE. The EIS will evaluate the potential environmental impacts associated with constructing, operating, monitoring, and eventually closing a repository at Yucca Mountain. Under Section 114(a)(1)(D) of the Nuclear Waste Policy Act of 1982, the EIS is not required to consider the need for a repository, alternatives to geologic disposal, or alternative sites. The EIS will evaluate a proposed repository designed to store up to 70,000 metric tons of spent nuclear fuel from commercial reactors and DOE-owned spent nuclear fuel and high-level radioactive waste. Based on public comments on the scope of EIS, the document will also include analyses for disposing of all projected spent nuclear fuel and other highly radioactive waste types that may be appropriate for disposal at Yucca Mountain.

The proposed action evaluates the potential environmental impacts associated with developing a repository, as well as transporting waste to a repository. Implementing alternatives have been developed by DOE to evaluate the range of potential environmental impacts that could occur. The implementing alternatives are defined to include a reasonable range of expected activities based on different approaches DOE could use to implement the proposed action. The draft EIS will be completed and issued to the public for review and comment by July 30, 1999. DOE will review and develop responses to all the public's comments. The final EIS will be published in August 2000 and subsequently will accompany the site recommendation report and the LA.

Environmental compliance. To comply with applicable environmental regulations, DOE will continue programs through site characterization to the eventual operation, closure, and decommissioning of the proposed Monitored Geologic Repository. These programs are associated with land use and reclamation; biological resources; air quality and weather; water quality and availability; cultural resources and values, with emphasis on the concerns of Native Americans; background and natural radiation monitoring; environmental justice; and management of hazardous and solid wastes and materials. The environmental compliance work includes preparing applications for required federal and state permits and monitoring to ensure that all permit objectives are met and compliance is maintained.

Site Recommendation

Under Section 114 of the Nuclear Waste Policy Act of 1982, a decision is required by the Secretary of Energy about whether to recommend the site to the President. If the site is recommended to the President by the Secretary, the President in turn, may recommend the site to Congress. If the President recommends the Yucca Mountain site to Congress and the designation is permitted to take effect under Section 115 of the Nuclear Waste Policy Act of 1982, as amended in 1987, the process of obtaining authorization from NRC to construct a repository will begin.

The site recommendation process begins with public hearings near Yucca Mountain to inform area residents that the Secretary is considering recommending the site, and to receive the residents' comments regarding the possible recommendation. These hearings are referred to as "consideration hearings" and would be announced by a notice of consideration. At the time this notice is published, DOE would also request the views and comments of the governors and legislatures of all of the states, and the governing bodies of any affected Indian tribes. If upon completion of these hearings and completion of site characterization, the Secretary decides to recommend approval of the site to the President, the Secretary will notify the governor and legislature of the State of Nevada. No sooner than 30 days later, the Secretary will submit to the President a recommendation that the President approve the site for the development of a repository.

Licensing

Preparation for licensing requires a number of activities, including developing licensing documents, resolving issues, and managing technical data and records.

Licensing documents. A number of licensing documents must be prepared, but the most important is the LA itself. It will present information on the site, repository design, and performance assessment to support DOE preclosure and postclosure safety cases. DOE will develop the LA in two phases, the working draft and the acceptance draft. The acceptance draft will follow the working draft and will contain the licensing safety case for the repository.

DOE is preparing a document to provide technical guidance on content to authors of the LA. In addition, DOE plans to prepare and submit topical reports to NRC on repository seismic design and postclosure nuclear criticality. The intent is to have NRC evaluate these topical reports and resolve any related licensing issues before DOE submits the LA. DOE must also prepare a documentary record for the LA, which consists of referenced citations, reports, evaluations, models, calculations, design packages, and technical data.

It should be noted that NRC announced plans to issue new, site-specific regulations for Yucca Mountain. The LA will address NRC requirements applicable to Yucca Mountain at the time the LA is submitted.

Issue resolution. DOE will address licensing issues by developing topical reports, interacting with NRC technical and management staff, and reviewing regulations as they apply to the licensing documents. The objectives of these activities are to obtain NRC prelicensing concurrence for unlicensed methods; request clarification or revision of NRC requirements and expectations; resolve technical and regulatory issues; address NRC comments on technical documents or activities; and identify, track, and complete commitments to NRC. A primary focus will be resolving the remaining 9 of the 10 key technical issues that the NRC staff has identified. These issues primarily address aspects of postclosure repository performance. The current status of these key technical issues is summarized in Section 4.3.3. Table 4-2 shows where in the VA each key technical issue is addressed. DOE is addressing these key technical issues through the site characterization process, including performance assessment, and various interactions with multiple participants. NRC staff will document resolution status of the issues in a series of issue resolution status reports.

Technical data management. To support the rigorous process of developing information for the site recommendation and the LA, DOE manages a massive amount of technical data in a number of databases. The quality of the data must be maintained in compliance with quality assurance requirements, DOE procedures, and federal records requirements. Managed data include spatial data providing location information for test activities, facilities, roads, and geological features; summaries and interpretations of geological and engineering test data; data from site investigations, and field and laboratory tests; characteristics of radioactive waste materials; and thermodynamic properties of chemical compounds. DOE maintains and will continue to build a master indexing system to allow data that are used in reports and analyses to be traced to their original sources.

Records management and licensing system network. DOE is upgrading its records information system to improve search and retrieval capabilities and to address changes in information management regulations and user needs. In addition, records processed before fiscal year 1996 are being converted to electronic images. The text is being converted to be computer searchable to allow the identification of records and computer retrieval of the image and text of licensing documents. Records generated since 1996 have been processed directly into this new system. DOE plans to develop an integrated electronic information system that will use available internet technology to allow NRC to access, through the NRC electronic data system, the DOE records management system, the online LA, and the technical database management system. This concept reflects the anticipated revision by NRC of 10 CFR 2, Subpart J.

SUPPORT ACTIVITIES

A number of activities are required to support the work described. These include field construction and operations activities as well as other support activities.

Field Construction and Operations Activities

The remaining site characterization work requires continuing construction and operation activities in the field. To meet this objective, DOE has established a field infrastructure that must be maintained consistent with the principles of sound stewardship of federal property and with site characterization objectives. The field facilities involve three general areas:

- Controlled land area, buildings and facilities used, and general infrastructure such as roads, utilities, and communication systems in and adjacent to Area 25 of the Nevada Test Site

- Surface test facilities, including boreholes and test excavations

- The subsurface Exploratory Studies Facility, surface support facilities, utilities, and underground openings created for site characterization purposes

Other Support Activities

In addition to field construction and operations, other activities necessary to support remaining work include information management, quality assurance, project management, security, institutional interactions, administrative services, and training support.

COSTS AND SCHEDULE

The estimated cost of completing the remaining work required to support an LA acceptable to NRC is $1.14 billion. These estimated costs are approximately evenly spread between October 1, 1998, and March 1, 2002.

Beginning with this VA, the key repository milestones are as follows:

- Publish a draft EIS in fiscal year 1999.
- Complete the final EIS in fiscal year 2000.
- If the site is suitable for development as a repository, complete the Secretary of Energy's site recommendation to the President, to be accompanied by the final EIS, in fiscal year 2001.
- If the President recommends the site to Congress, and the designation becomes effective, submit an LA to NRC in 2002.

1. INTRODUCTION

Volume 4 provides the DOE plan and cost estimate for the remaining work necessary to proceed from completing this VA to submitting an LA to NRC. This work includes preparing an EIS and evaluating the suitability of the site. Both items are necessary components of the documentation required to support a decision in 2001 by the Secretary of Energy on whether or not to recommend that the President approve the site for development as a repository. If the President recommends the site to Congress and the site designation becomes effective, then DOE will submit the LA to NRC in 2002 for authorization to construct the repository.

The work described in Volume 4 constitutes the last step in the characterization of the Yucca Mountain site and the design and evaluation of the performance of a repository system in the geologic setting of this site. The plans in this volume for the next 4 years' work are based on the results of the previous 15 years' work, as reported in Volumes 1, 2, and 3 of this VA. Volume 1 summarizes what DOE has learned to date about the Yucca Mountain site. Volume 2 describes the current, reference repository design, several design options that might enhance the performance of the reference design, and several alternative designs that represent substantial departures from the reference design. Volume 2 also summarizes the results of tests of candidate materials for waste packages and for support of the tunnels into which waste would be emplaced. Volume 3 provides the results of the latest performance assessments undertaken to evaluate the performance of the design in the geologic setting of Yucca Mountain. The results described in Volumes 1, 2, and 3 provide the basis for identifying and prioritizing the work described in this volume. DOE believes that the planned work, together with the results of previous work, will be sufficient to support a site suitability evaluation for site recommendation and, if the site is recommended and designated, a defensible LA.

Volume 4 is divided into seven sections. Section 2 presents a rationale and summary for the technical work to be done to develop the preclosure and postclosure safety cases that will support the compliance evaluations required for the evaluation of site suitability and for licensing. Section 2 also describes other necessary technical work, including that needed to support design decisions and development of the necessary design information. Section 3 presents a more detailed description of the technical work required to address the issues identified in Section 2. Section 3 also describes activities that will continue after submittal of the site recommendation and the LA. Examples include the drift scale heater test in the Exploratory Studies Facility (Section 3.1.4.3) and long-term waste package corrosion testing (Section 3.2.2.9). Section 4 discusses the statutory and regulatory framework for site recommendation and submittal of an LA, and describes the activities and documentation that must be completed to achieve these milestones, including the development of an EIS. Section 5 describes the numerous activities required to support program milestones, including support for completing the testing program, continuing tests as part of the performance confirmation program, and managing information and records to support regulatory and legal review. Sections 6 and 7 provide cost and schedule information for the activities planned.

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2. RATIONALE FOR TECHNICAL WORK NEEDED TO COMPLETE THE LICENSE APPLICATION

Investigation of the Yucca Mountain site has been based on a progressively improving picture of how the site would support a repository system. As information on the site has increased, that picture has developed, the notion of what is critical to the system has sharpened, and the work plans have concentrated on a decreasing number of key uncertainties. The repository investigations have provided substantial understanding of the site, preliminary design information, and performance assessments of the type needed for decisions regarding the site recommendation and the LA.

However, additional technical work is needed. There are still several issues for postclosure performance and preclosure safety that need to be addressed. In addition, a full range of design alternatives that might enhance confidence in system performance have yet to be evaluated. Technical work must still be conducted to support critical decisions regarding the degree of design margin and defense in depth needed to mitigate uncertainties, controls to address nuclear criticality, and other issues.

2.1 OVERVIEW OF RATIONALE FOR NEEDED TECHNICAL WORK

This section gives the rationale for the remaining technical work. Work will be done to:

- Complete the postclosure safety case
- Support the preclosure safety case
- Support design decisions

The technical work identified here focuses on key milestones. The decision on site recommendation is scheduled for July 2001. Submittal of the LA is scheduled for March 2002. The required technical information must be provided in advance of these actions to support full integration of site information, design, and performance assessment. In particular, information to support initial selection of the design must be available by May 1999, and information to support proposed selection of the reference design must be available by November 2000 so that it may be incorporated into the site recommendation and the LA. Information to support selecting process models must be provided by August 1999, in time to support performance assessment evaluations that will begin then. Information to support a draft of the site recommendation and the LA must be available by January 2000.

2.2 RATIONALE FOR TECHNICAL WORK NEEDED TO COMPLETE THE POSTCLOSURE SAFETY CASE

Based on the site characterization results presented in Volume 1, the reference repository design and potential design enhancements described in Volume 2, and the performance assessment results presented in Volume 3, DOE believes that work should proceed as planned. The postclosure safety case needed for site recommendation and the LA is nearly complete. A few issues remain to be addressed, but these are well defined and the work to address them is well understood. Design flexibility exists to enhance the performance of a repository, if necessary, and a process is underway to evaluate design alternatives and options to select an appropriate reference design for the LA. The remaining work can be completed on time to support both the site recommendation and the LA. The following discussion summarizes current information, assessments of its adequacy, information that is yet needed, and the priorities of the remaining testing, design, and performance assessment work.

Section 2.2.1 reviews the current status of the postclosure safety case and summarizes the general information needed for quantitative estimates of expected postclosure performance, assessment of the degree of design margin and defense in depth provided by the system, consideration of disruptive processes and events, insights from natural and man-made analogs, and a performance confirmation plan. This section establishes the general framework for the remaining information needed to complete the postclosure safety case.

Section 2.2.2 summarizes the repository safety strategy (DOE 1998b), which is the plan to complete the postclosure safety case. The document identifies four key attributes of an unsaturated zone repository system that are related to meeting postclosure performance objectives. Within these key attributes, the strategy identifies principal factors of greatest importance to postclosure performance.

Section 2.2.3 describes the 19 principal factors important to postclosure performance of the reference design. This section evaluates the potential significance of each factor.

Section 2.2.4 gives the priorities for technical work that will address information needs associated with each of the 19 principal factors. The prioritization is based on:

- The level of current confidence that has been obtained for each factor, based on the technical work performed to date
- The corresponding confidence goal for each factor, which is the confidence level considered to be feasible and desirable at the time of the LA

The highest-priority principal factors emerging from this analysis are seepage into the emplacement drifts, integrity of the corrosion-resistant waste package barrier, and transport properties of the unsaturated zone.

Section 2.2.5 discusses VA reference design options that have been examined in performance assessment sensitivity studies. These options indicate ways that the performance of the repository system may be improved. The section discusses the potential enhancements that are possible and the current issues that need to be addressed.

Section 2.2.6 describes the technical work planned to complete the postclosure safety case. Work will address questions about the performance of engineered barriers (i.e., longevity of particular materials and feasibility of specific designs) and uncertainties in site characteristics (e.g., the rate of water seepage into underground openings where waste is emplaced).

2.2.1 The Postclosure Safety Case

As indicated in the preceding paragraphs, the postclosure safety case is the set of data and analyses that will be used to provide reasonable assurance that a repository at Yucca Mountain will protect the health and safety of the public and will satisfy regulatory objectives and criteria for postclosure performance. As stated in the applicable 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 the postclosure safety case, which are intended to provide this reasonable assurance, 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 following paragraphs discuss the current status of each of these elements and describe the nature of the remaining work to complete the postclosure safety case in that area.

2.2.1.1 First Element of the Postclosure Safety Case–Assessment of Expected Postclosure Performance and Supporting Evidence

Current Status of the Case. The current assessment of expected performance is summarized in Volume 3. That assessment reveals that the vast majority of the radionuclides in the repository at the Yucca Mountain site are not mobile: either they are insoluble or they sorb strongly to minerals and materials in the repository and cannot move out of the repository (Volume 3, Section 4.2.4). A very 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 can be kept from contacting the waste.

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 a semiarid region and the nature of the site limit the amount of water that can reach the repository to a small fraction of the rainfall at the ground surface (averaging approximately 5 percent under present climatic conditions). 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 feature, the waste packages would be protected from changes in conditions at the surface, while still being kept well away from groundwater. The site would, therefore, provide a predictable environment for which engineered barriers can be designed to further limit the exposure of waste to water.

Performance assessment studies show that in addition to the natural barriers, engineered barriers would keep water away from the waste. These studies indicate, for example, that the highly corrosion-resistant inner barrier and the thick steel outer barrier of the waste package reference design for this VA would each provide effective barriers against water for different conditions occurring during the postclosure performance period. Although there are some issues that must still be addressed, the estimates indicate that robust waste packages could be designed to remain intact for thousands of years in the repository environment. The studies also indicate that the spent nuclear 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.

These conclusions follow in part from performance assessment analyses of expected behavior of the repository system and sensitivity studies of the factors contributing to this behavior. Volume 3 presents the most recent set of DOE performance assessment analyses. These analyses are based on assessment of all available data for processes and conditions that might occur and the likelihood of occurrence. Models of these processes and conditions are used to estimate the performance of the system in numerical simulations. These simulations provide one important method of assessing postclosure safety of the repository system and are useful in assessing the relative contributions of various factors. However, the results of the computational models should not be interpreted as an accurate prediction of the actual performance of the repository. As described in Volume 3, uncertainty in the detailed results of performance assessment remains because of spatial and temporal variability in both site conditions and the complexity of coupled physical processes that will operate in the repository environment.

Information Needs and Technical Work Needed to Complete the Case. Characterization work at the site has significantly reduced the number and magnitude of the uncertainties important to postclosure performance; however, uncertainties remain, including uncertainties in the variability of the flow of water at the site and the transport of radionuclides that might be released from breached waste packages. These uncertainties will be reduced through additional technical work on information needs that have been identified from analysis of the principal factors of the repository safety strategy. These information needs are expressed as items of site investigation, experimental work, analysis of data, or refinement of predictive simulations that are important to the safety case. This work will reduce uncertainty and will improve estimates of repository performance. The information needs are summarized next and are consistent with a systematic prioritization of the principal factors of postclosure performance, presented later in this section.

- **Percolation and Drift Seepage.** Additional information on the range of flow rates and the location of flow pathways in the unsaturated zone will improve estimates of water seeping into drifts. In addition, better understanding the physics of seepage flow and the changes that may result from thermally driven coupled processes and the eventual collapse of the drift openings will support more accurate predictions of seepage under future conditions. These information needs will be addressed by work to continue tests of water movement in the unsaturated zone and seepage into openings in the Exploratory Studies Facility. Chemical and isotopic analysis of samples from the underground facility will continue to provide information on the distribution of flow in the unsaturated zone. The effects of heating are being investigated in a drift-scale thermal test that is presently underway and by laboratory experiments that will support models for predicting the effects of coupled processes over much longer periods. The end result of work in this area will be refined models for percolation, seepage, coupled processes, and introduced material interactions that can affect the amount of water that contacts the waste packages.

- **Effects of Heat and Excavation on Flow.** Additional information is needed to refine the material properties and physics used in models for the host rock and the engineered barriers, for more accurately predicting the movement of water during the thermal period. Information is also needed to improve the basis for predicting permanent changes in material properties that could affect the movement of water after the repository system has cooled down. Information needs pertaining to chemical and mechanical coupled processes and the movement of water during the thermal period will be met by ongoing and planned field thermal tests. Laboratory experiments will provide additional information about chemical processes that proceed more slowly and cannot be readily observed on the time scale of a few years. The drift-scale models used to describe the movement of heat, water, and chemical components will be updated.

- **Water Dripping Onto Waste Packages.** Movement of water from the drift wall to the waste packages will depend on the nature of seepage and the physics of flow within the drift. Additional information is needed to refine the model for partitioning seepage flow between water that drips onto the waste packages, and water that does not. For the VA reference design, process models generated for TSPA indicate that the distribution of drips in drift openings may be nonuniform and that models focusing flow onto the waste packages may be overly conservative. Work will emphasize engineered measures to control water dripping onto the waste packages and will be based on both predictive modeling and experiments (Section 3.2.1.12). The engineered barrier system will be tested in laboratory settings and in larger scale proof of principle testing arrangements. These needs will also be addressed by field testing of seepage into openings, which will compile observations of dripping behavior for a range of percolation conditions. Predictive models for dripping behavior will also be based on analysis of fracturing in the host rock.

- **Chemistry of Water on Waste Packages.** Chemistry in the near-field environment is a factor in determining corrosion rates for material comprising the engineered barrier, particularly those for the waste packages. The near-field geochemical environment will be affected by the host rock, introduced materials such as concrete and steel, and the composition of the gas-phase circulation. Additional information is needed on the nature and rates of physical and chemical interaction between the introduced materials and water in the near-field environment. In addition, more must be learned about the influence that mountain-scale, gas-phase processes have on near-field chemistry. These information needs will be addressed through experimental evaluation of introduced materials and by simulation of gas-phase processes using models tested against observations from the field thermal tests. Waste package corrosion may be strongly influenced by micro-environments on the package surface. These micro-environments are associated with films, pits, or crevices that may form as a result of abiotic or microbially mediated processes. These mechanisms will be investigated in connection with long-term material testing and waste package design.

- **Integrity of the Inner Corrosion-Resistant Waste Package Barrier.** Additional information is needed to describe the modes of corrosion and the long-term metallurgical stability of the waste package barrier materials. Studies will include observing the rate of formation and thickness of passivation layers that provide corrosion resistance. Additional data on corrosion processes are needed to refine and support models for localized corrosion and phase stability. Work will address these needs by continuing to test materials under a wide range of environmental conditions, investigating material structure, continuing to study microbially induced corrosion, and refining predictive models.

- **Integrity of Spent Nuclear Fuel Cladding.** The frequency of, and mechanisms for, failure of irradiated Zircaloy cladding are important uncertainties in assessing cladding performance. Performance data for spent nuclear fuel in dry cask storage are needed to establish the likely condition of the spent nuclear fuel inventory at the time of disposal and to refine current models for cladding failure in the repository. These needs will be addressed by ongoing tests to assess Zircaloy corrosion and planned studies of early fuel rod failures in dry storage.

- **Formation of Radionuclide-Bearing Colloids.** Additional information is needed to refine the wide range of possible colloidal transport behavior that is presently assumed in TSPA, particularly with respect to the stability of radiocolloids, the reversibility of radionuclide sorption to colloidal particles, and the mobility of colloids in groundwater. Experimental data are needed for sorption and desorption behavior and the relationship of solubility limits to colloid formation needs to be further defined. Laboratory studies of colloid formation and stability and their interactions with radionuclides will address these needs.

- **Transport Through the Unsaturated Zone.** To refine the current model for radionuclide transport in the unsaturated zone, additional data are needed to support the treatment given to several processes. Measurements are needed to support evaluating the extent to which dilution of fracture-borne radionuclides occurs from fracture-matrix water interaction and other processes. Additional measurements are needed to support modeling retardation by matrix diffusion under unsaturated conditions and the possible effects of coupled processes on fracture-matrix interaction. Additional data on colloid transport are needed, particularly for non-reversible, competitive sorption of radionuclides to colloids and colloid filtration effects. In addition, the model of unsaturated zone transport needs to better represent fault zones and the spatial variability of transport properties. These needs will be met by laboratory testing, by the ongoing field geochemistry test at Busted Butte, and by experiments in the Exploratory Studies Facility cross drift (which will examine the transport of multiple tracers in the host rock). Radionuclide transport data from former DOE weapons facilities will be examined, particularly with respect to colloid transport. The unsaturated zone flow and transport model will be updated to incorporate new geologic constraint data and results from field tests.

- **Flow and Transport in the Saturated Zone.** Additional information concerning the hydrogeologic framework and the treatment given to several transport processes not included in the present model is needed to refine the saturated zone flow and transport model. Transport model measurements are needed to support incorporating dispersion, chemical retardation, and matrix diffusion resulting from fracture-matrix interaction. Additional measurements of hydraulic and transport parameters are needed from locations and depths that have not yet been tested. Hydrochemical data are needed from existing boreholes and new boreholes constructed in cooperation with Nye County, and these data need to be incorporated into the conceptual and predictive models for saturated zone processes. The refined process models will then be adapted for use in performance assessment. Additional effort is also needed to develop the regional flow model, in cooperation with the U.S. Geological Survey (USGS) and DOE/Nevada Operations, to ensure that a consistent model is available for groundwater monitoring and water supply applications throughout the region.

These information needs are also addressed in the prioritization process, described in Section 2.2.4, in the context of goals for increasing confidence in principal factors of postclosure performance. In addition there are other, lower priority information needs that are addressed in the prioritization or identified in the work plans in Sections 3.1 and 3.2.

2.2.1.2 Second Element of the Postclosure Safety Case–Design Margin and Defense in Depth

Current Status of the Case. Because there will always be uncertainties in estimates of repository performance over many thousands of years, DOE is not relying solely on these estimates to demonstrate reasonable assurance that postclosure performance standards will be met. Design margin and defense in depth provide important additional assurance.

"Design margin" refers to the standard engineering practice of including margins of safety in specifications for engineered components to account for uncertainty in the conditions to which the components will be subjected and for variability in the properties of component materials. "Defense in depth" is the term used here to describe the property of a system of multiple barriers to mitigate uncertainties in conditions, processes, and events by employing barriers that are redundant and independent, such that failure in any one barrier does not result in failure of the entire system.

Current performance assessments indicate that the robust waste packages of the reference design, in the environment provided by the unsaturated zone at Yucca Mountain, are expected to keep water away from waste for thousands of years. The outer and inner waste package layers have different corrosion modes and are an example of defense in depth provided by the reference design. Other barriers, such as drip shields, backfill, and ceramic coatings to the waste package, could be added to the reference design to provide additional design margin and defense in depth. Drip shields, for example, could be constructed from different corrosion resistant materials than the waste packages and, thereby, mitigate the consequences of waste package corrosion.

Site characterization and laboratory testing have resulted in a reasonable understanding of the environments in which the engineered barriers would perform. Based on this understanding, DOE will analyze the design margin and defense in depth provided by the reference design, design, options, and alternative designs.

Information Needs and Technical Work Needed to Complete the Case. Design margin and defense in depth will be analyzed in three steps. First, the conditions and scenarios to be considered in the evaluation will be defined. These cases will represent extreme conditions, some of which are unanticipated but still sufficiently credible to warrant consideration. Some of these conditions and scenarios may be prescribed by regulations or identified in regulatory guidance documents.

The second step will be to assess the performance of the system assuming the conditions and scenarios chosen in the first step. These calculations will focus on key radionuclides (e.g., those that are relatively mobile, long-lived, and could contribute significantly to dose). The analyses will focus on assessing the key functions of the repository system: limiting water contact with the waste packages, achieving long waste package lifetime, limiting the rate of release of radionuclides from breached waste packages, and reducing radionuclide concentration during transport from the waste packages.

The third step will be a series of calculations in which each critical component of the repository system is postulated to fail prematurely. The results of these analyses will not be predictions of actual performance, but will serve to compare the ability of alternative designs to isolate waste under both anticipated and unanticipated conditions and to demonstrate defense in depth.

The work to address design options of the VA reference design is summarized in Section 2.2.5. The evaluation of alternative designs in terms of their design margin and contribution to defense in depth will be conducted as part of a comprehensive assessment of design alternatives and options. This comprehensive evaluation, as discussed in Section 2.4.3, will be conducted in 1999. In addition to design margin and defense in depth, the design alternatives evaluation will address other objectives including defensibility of the postclosure safety case, flexibility of design, schedule, cost, and preclosure safety.

Information needs for this element are expressed as items of experimental work, analysis of data, or refinement of predictive capabilities that are needed to predict the performance of engineered barriers. The following discussion describes the information needed to evaluate the design margin and defense in depth that could be provided by the VA design options. The design options are discussed further in Section 2.2.5 and in Volume 2, Section 5.3.

- **Drip Shield and Backfill.** The drip shield and backfill options would prevent seepage water from contacting waste packages by imposing an impervious barrier embedded within, or covered by, a protective layer of drift backfill. To simulate the performance of this combination, the flow properties of candidate backfill materials are needed along with an assessment of long-term changes in backfill. In addition, the characteristics of drift seepage and drainage are needed for performance simulation. Analyses of the engineering feasibility of barrier design and emplacement are needed. To establish the longevity of the drip shield, additional information on material degradation processes and rates is needed. These needs will be met by planned measurements of the thermal, hydrologic, and transport properties of backfill at unsaturated conditions. Candidate backfill materials will also be subjected to chemical alteration under hydrothermal conditions similar to what might occur in the near-field environment. Engineering analyses will be performed for the comprehensive evaluation of design alternatives. Ongoing studies of candidate drip shield material properties and longevity will continue with the objective to support and confirm the material selection decisions made in the design alternatives evaluation. Engineered barrier concepts will be tested in scaled, prototype models to confirm engineering studies and numerical simulations (Section 3.2.1.12).

- **Ceramic Waste Package Coating.** Additional information is needed to support assessments of the longevity of ceramic coatings on the waste package, drip shield, or other engineered barriers. The stability of ceramics against phase transitions, and the long-term continuity of the coating, need to be further evaluated. The reliability of spray coating methods for application and the long-term effects of the metal substrate also need additional work. Information is also needed on the effectiveness of backfill to prevent mechanical degradation caused by rockfall. Information on ceramic material properties, such as permeability, density, strength, adhesive strength, and the responses to thermal loading and handling loads, will be met by reviewing industrial experience and conducting further tests as necessary. Corrosion tests will be performed as functions of ceramic thickness, structure, and composition. Natural or man-made analogs will be used to the extent practicable for assessing the longevity of ceramics or other man-made materials.

Information needs for design margin and defense in depth analyses may change as a result of the systematic evaluation of design alternatives, which could change the engineered barrier system relative to the VA reference case and design options.

2.2.1.3 Third Element of the Postclosure Safety Case–Consideration of Disruptive Processes and Events

Current Status of the Case. The postclosure safety case will explicitly consider processes and events with the potential to 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.** Yucca Mountain is located in the southern Great Basin, a region of slow but ongoing tectonic deformation that is associated with earthquakes. Yucca Mountain itself is a tilted block of rock that is bounded by geologic faults. A magnitude 5.6 earthquake occurred about 20 km (12 miles) away from the site in 1992.

  DOE has completed an analysis of the earthquake ground motion and fault displacement hazard at the Yucca Mountain site (Volume 1, Section 2.2.7.2). Work to determine site-specific soil and rock properties (Section 3.1.1.4) will enable determination of final ground motion design values. DOE expects the resulting ground motion design bases can be accommodated using standard practices used in nuclear design and construction. The fault displacement hazard analysis indicates the fault displacement hazard is low and can be accommodated by avoiding known faults in the layout of repository facilities (Volume 1, Section 2.2.7.2). Quantitative estimates of the effects of earthquakes on postclosure performance show no significant impact on performance (Volume 3, Section 4.4.3.3). This work suggests that this issue can be addressed by current information.

- **Volcanism.** The large-volume, silica-rich volcanism that created the rock of Yucca Mountain ceased in the region about 7.5 million years ago. This type of volcanism was succeeded by relatively low volume basaltic volcanism, as evidenced by the cinder cones near Yucca Mountain. The geologic record indicates that the rate of basaltic volcanic activity is declining, with the last episode occurring about 75,000 years ago. Statistical analyses of the pattern and rate of cinder cone activity around Yucca Mountain over the last million years indicates that the average risk of a volcanic disruption at the site in any given year is approximately 1 in 70 million (Volume 1, Section 2.2.7.1). Performance assessments that consider several extreme volcanism scenarios indicate that most scenarios would have little impact on postclosure performance (Volume 3, Section 4.4.2.4). This work suggests that this issue can also be addressed with available 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 peak dose rates would increase in a scenario in which exploratory drilling penetrates a waste package and waste is carried down to the water table (Volume 3, Section 4.4.5.2). However, mineral resource assessments indicate that the Yucca Mountain site does not exhibit characteristics that make it particularly attractive for exploration (Volume 1, Section 2.2.7.3). Therefore, although an increase in radiation exposure cannot be precluded for some human intrusion scenarios, available information strongly suggests that the probability of these scenarios is very small. 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

(Volume 2, Section 5.1.3.1). Analyses have been conducted to address the possibility of a criticality event outside the waste package in the event of a breach of the waste package. These analyses indicate that the concentrations of dissolved radionuclides will be too low to result in the accumulation of a critical mass outside the waste package in fewer than a million years (Volume 3, Section 4.4.4.3). However, the potential for transport and accumulation of these radionuclides by way of colloids has not yet been completely analyzed. Consequently, additional technical work is needed to address this possibility.

Information Needs and Technical 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 methods that will be used to evaluate criticality, and consolidating and documenting current information. Although more information on site-specific soil and rock properties is needed to finish development of seismic design requirements, it appears that current information is adequate to address postclosure performance issues (Volume 1, Section 2.2.7.2). Likewise, current information is adequate for evaluating the postclosure effects of volcanism and inadvertent human intrusion for natural resource extraction (Volume 1, Section 2.2.7.3). Technical work regarding postclosure nuclear criticality is nearly complete, although information is needed to address the following aspects of radionuclide transport and criticality analysis:

- Formation and stability of colloids bearing fissionable radionuclides
- Transport of colloids in the near-field environment and altered zone
- Final methods for addressing criticality in the regulatory arena

These needs will be addressed by planned laboratory work to evaluate colloid formation from spent nuclear fuel interaction with water and to investigate colloid interaction with, and mobility through, introduced materials in the near-field environment. In addition, an improved general understanding of colloid mobility in the unsaturated zone will result from the field transport test at Busted Butte (see Volume 1, Appendix C). Additional work planned to address colloidal transport of radionuclides is discussed in Section 3.1.2.1 of this volume, and work to address the remaining questions on criticality methodology is discussed in Section 2.4.1.

2.2.1.4 Fourth Element of the Postclosure Safety Case–Insights from Natural and Man-Made Analogs

Current Status of the Case. Relevant information about the future waste isolation performance of the site 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 analogs. No analog exists for the complete Yucca Mountain repository system; however, there are sites that can provide information on processes and conditions relevant to a repository system at the site, which cannot be studied in other ways due to the long time periods or large spatial scales involved. An important advantage of analogs is that processes that have been ongoing for very long periods can be studied. Analogs provide the only means of looking at long time periods and large spatial scales. Analogs are identified in 10 CFR 60 as an additional means to provide assurance that safety standards for the repository will be met.

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 (Volume 1, Section 2.2.5.3). Similarly, man-made analogs may provide relevant information over time scales and distances that are not reproducible in a laboratory setting or in limited-duration field studies. For example, examination of ancient ceramic artifacts could provide useful information regarding the stability of candidate ceramic materials proposed for use as coatings on drip shields or waste packages. Natural analogs have significant limitations, including the incomplete and heterogeneous geologic record, uncertainty in characterizing the past conditions, partial or imperfect analogy to repository conditions, and divergent interpretations of geologic data. Furthermore, data from analog studies are generally independent of site characterization and modeling for a Yucca Mountain repository and may provide independent verification of the models used for postclosure performance assessment. Because of this consideration care must be exercised in the selection of appropriate analogs.

Information Needs and Technical 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. Information is needed for natural analogs to address the following aspects of repository performance and design:

- Seepage into drifts
- Long-term stability of metals in an oxidizing environment
- Solid species chemistry of radionuclides, including uraniferous minerals and coprecipitation (analogs to waste form degradation)
- Long-term stability of concretes
- Transport of radionuclides in unsaturated, oxidizing conditions
- Thermal-hydrological-chemical processes
- Long-term mechanical stability of excavations under thermal conditions
- Retardation and diffusion in backfill

This information will complement the short-term laboratory tests and field tests conducted at Yucca Mountain.

To prepare for the site recommendation and LA decisions, a comprehensive review and summary of analog information relevant to performance of a Yucca Mountain repository will be compiled. This compilation will build confidence in process models, TSPA, and repository design. This review will include environmental analog sites that have been characterized for inference of rainfall and infiltration from remnants of the ice-age plant community at Yucca Mountain. It will also include analogs for radionuclide solubility and geochemical processes that affect transport, which 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 nuclear fuel radionuclides. Geothermal areas, which provide a means to exercise numerical models of coupled thermal, hydrologic, and chemical processes, will be included. Transport of radionuclides at analog sites and man-made analogs of materials that may find application in the engineered barrier system, will be also be included. Technical work to assess natural and man-made analogs is summarized in Sections 3.1 and 3.2.1.

2.2.1.5 Fifth Element of the Postclosure Safety Case–A Performance Confirmation Plan

Current Status of the Case. The final element of the postclosure safety case is a performance confirmation plan, the plan for long-term testing and monitoring. Performance confirmation provides the means to address inherent limitations and uncertainties associated with performance and design analyses that cannot be completely addressed in short-term testing and site characterization. Performance confirmation begins with the documentation of current conditions, which is performed during site characterization, and continues until the repository is permanently closed. The information collected during performance confirmation will be particularly relevant to long-term analyses of the repository. The increased understanding and confidence derived from long-term testing and observation will inform future decision makers, who will need to determine when to apply to NRC for authorization to close the repository.

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

- Confirm that subsurface conditions encountered during construction, waste emplacement operations, and monitoring are within the ranges assumed in the LA

- Confirm that natural and engineered systems and components are functioning as intended and anticipated

- Evaluate compliance with NRC postclosure performance requirements

- Evaluate the repository readiness for permanent closure

A performance confirmation plan has been developed (CRWMS M&O 1997d) 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 will depend on the LA reference design that is selected in 1999 and any corresponding changes to the postclosure safety case. The current plan is therefore preliminary and provides an outline of the areas to be addressed by testing and monitoring during repository construction, waste emplacement, and the long-term monitoring phase of repository operations.

Information Needs and Technical Work Needed to Complete the Case. Specific tests will be defined and the performance confirmation plan will be revised once the design for the LA has been chosen. The information needs for this element of the safety case therefore depend on the LA reference design. The revised performance confirmation plan will define the activities necessary to implement the requirements of 10 CFR Part 60, Subpart F, for the LA reference design. It will specify monitoring, testing, and analysis activities to be conducted to provide additional assurance that the postclosure performance objectives will be met.

The performance confirmation program defined in the revised 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 which is available for incorporation in the LA.

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 LA, the mathematical models formulated for these processes, the software models that have been developed to simulate these processes, and the parameters required for these models. The plan will describe the process of integrating the additional information obtained after submittal of the LA into these models.

2.2.2 Repository Safety Strategy

2.2.2.1 Role of the Repository Safety Strategy

The general information needs summarized in the preceding paragraphs provide the framework for defining the technical work to be completed before site recommendation and the LA. The information needed to complete the postclosure safety case for these purposes is identified in the repository safety strategy (DOE 1998b). This document integrates the needs from site characterization, repository system design, and performance assessment within the context of four key attributes of the repository 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

The strategy addresses these attributes in two ways. First, it identifies the principal factors important to each attribute and assesses the importance of the uncertainties in them and second, it discusses evaluation of design alternatives and options that could improve performance or increase confidence in postclosure safety. DOE has identified 19 factors, which reflect the features of the natural geologic system at Yucca Mountain and the reference design considered here, that are of greatest importance to postclosure performance. These factors are listed in Table 2-1 and are discussed in detail in Section 2.2.3.

Section 2.2.4 summarizes the assessment of the status of these factors. This assessment concludes that the information needed about these factors for the postclosure safety case is nearly complete. It identifies the few key issues that remain to be addressed and estimates the level of confidence needed for a successful LA.

A key element of this assessment of the principal factors was a review of the current confidence in the representations of these factors in TSPAs. Coupled with this review was a detailed review by outside experts to identify limitations in the models used for the TSPAs (Whipple et al. 1998). That review concluded that DOE can address current limitations in these models by specific tests and analyses and by invoking design options to mitigate their effects. The second element of the assessment was an evaluation of the feasibility of conducting the needed work according to the schedule required to support site recommendation and the LA. That evaluation estimated "confidence goals" (based on needed levels of confidence and feasibility of attaining those levels) to prioritize that work and concluded that the needed work can be conducted.

The second area addressed by the repository safety strategy (DOE 1998b) is an evaluation of design alternatives and options that, within the context of the four key system attributes, might improve performance or otherwise increase confidence in postclosure safety of the repository system. Design options for the reference design considered here are discussed in Section 2.2.5. The repository safety strategy points to a comprehensive evaluation of potential design features, as well as alternatives to the reference design considered here, as they might enhance performance margin and defense in depth. That evaluation is ongoing and will be completed for initial design selection and to support the postclosure safety case for the site recommendation and LA decisions.

Table 2-1. Key Attributes of the Repository Safety Strategy and Principal Factors Affecting Postclosure Performance for the Viability Assessment Reference Design

Attributes of the Repository Safety Strategy	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 packages
	Humidity and temperature at waste packages
Long waste package lifetime	Chemistry of water on waste packages
	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 packages
	Integrity of spent nuclear fuel cladding
	Dissolution of spent nuclear fuel and glass waste forms
	Neptunium solubility
	Formation and transport of radionuclide-bearing colloids
	Transport through and out of the engineered barrier system
Radionuclide concentration reduction during transport from waste packages	Transport in the unsaturated zone
	Flow and transport in the saturated zone
	Dilution from pumping
	Biosphere transport and uptake

DOE believes that the combination of current and forthcoming information to address the 19 principal factors and the design alternatives and options will be sufficient to complete the postclosure safety case. There is a plan in place to obtain this information and, if work is completed according to this plan, the resulting information should support a successful LA. The details of the assessments leading to this conclusion are provided in the following sections.

2.2.2.2 Evolution of the Repository Safety Strategy

The repository safety strategy has evolved since the original strategy was set down in the *Site Characterization Plan: Yucca Mountain Site, Nevada Research and Development Area* (DOE 1988). This section discusses that evolution.

The strategy presented in the initial draft of the waste containment and isolation strategy prepared in 1996 updated the strategy in the site characterization plan (DOE 1988) for demonstrating safety of the repository system. The updated strategy incorporated new site characterization information, new repository and waste package designs, more realistic performance predictions, and changing regulatory considerations (i.e., risk-based or dose-based standard). The waste containment and isolation strategy identified the following five key system attributes:

- Rate of water seepage into the repository
- Waste package lifetime (containment)
- Rate of release of radionuclides from breached waste packages
- Radionuclide transport through engineered and natural barriers
- Dilution in the saturated zone below the repository

Specific working hypotheses were developed for each attribute to guide the testing and analysis work to address the issues regarding postclosure system performance. The hypotheses provided a basis for organizing, managing, and explaining the rationale for that work. Twelve hypotheses were originally delineated for the key attributes.

The waste containment and isolation strategy also addressed potential disruptions to the system that could conceivably result in the release of radionuclides directly to the accessible environment or otherwise affect the system performance (e.g., by adversely affecting characteristics of the system). Tectonics, seismicity, and volcanism were identified as potentially disruptive events. Three hypotheses associated with these events were also identified.

In 1998 the strategy was revised and issued under the title of *Repository Safety Strategy: U.S. Department of Energy's Strategy to Protect Public Health and Safety After Closure of a Yucca Mountain Repository* (DOE 1998d). The underlying concepts were not changed, but the key attributes were consolidated from five to four:

- 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

In addition, the hypotheses were increased from 15 to 18. No hypotheses were deleted, although some were combined. New hypotheses were added, and some hypotheses were restated. As in the previous strategy, the hypotheses were identified as a means of summarizing the remaining issues associated with the key attributes and disruptive processes and events. Since their development, work has been conducted to address the hypotheses, and this has resulted in a clear focus of the remaining issues. The results of such work have led to the identification of "principal factors," factors that have a significant role in assessments of postclosure performance of the site.

This transition from hypotheses to principal factors was concluded with issuance of Revision 2 of the repository safety strategy (DOE 1998b). This revision focuses on the principal factors as a tool for identifying those processes of the engineered and natural features that have a significant role in the postclosure performance of the site, and as a means of prioritizing the technical work to be done for the postclosure safety. The revised strategy also focuses on specific design options and their potential effect on postclosure performance if they were to be used in the design. The evaluation of these options also helps to prioritize the work to be done.

As the principal factors and design options and alternatives are used as a means of refining the performance allocated to the various natural and engineered features, the composition and number of principal factors may change. For example, the design alternatives evaluation process (see Section 2.4.3) planned will make use of the principal factors, and their importance to postclosure performance, in evolving the repository design. This will likely result in a change in the set of principal factors.

2.2.3 Principal Factors of Postclosure Performance

This section summarizes the importance of each of the 19 principal factors in Table 2-1 and the sources of technical information currently available to address each one. Discussion of the technical work planned to further develop each principal factor for site recommendation and LA decisions, and the prioritization that is the basis for selecting that work, is provided in Section 2.2.4. Specific work plans describing the prioritized work in more detail are presented in Section 3.

The principal factors affecting postclosure performance have been identified in performance assessment sensitivity studies based on the VA reference design. The most important critical factors of postclosure performance are summarized in Section 6.2 of Volume 3:

> ". . . the vast majority of radionuclides in the waste is immobile and never leaves the repository even if in contact with water. A small number of radionuclides, notably technetium-99, iodine-129, neptunium-237, and those radionuclides transported by colloids, are sufficiently mobile under some conditions that they could reach the biosphere downgradient from the repository. Therefore, these analyses indicate that the most important factors for system performance over time are the amount of water likely to contact the waste packages and the amount of waste exposed to that water. Consequently, long lived waste packages and engineered barriers or other factors that are effective in further limiting the contact of water with the waste will also be highly important to performance."

These sensitivities indicate that the principal factors contributing to peak dose rate, which are most relevant to the highly mobile radionuclides technetium-99 and iodine-129, are those which are related directly to the ability of the system to limit any contact of water with the waste, and reduction of concentration during transport away from the waste packages. The principal factors for neptunium-237, and radionuclides that may be transported by colloids, also include the factors that affect the amount of water contacting the waste packages, the lifetime of the waste packages, and the rates of release of these radionuclides from breached waste packages. Combining the sensitivities associated with all radionuclides, the principal factors are as shown in Table 2-1. A brief discussion of each principal factor follows.

Precipitation and 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 (Volume 1, Section 2.2.3.2). 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. Overall, performance assessment sensitivity analyses indicate that this factor is moderately important to performance (Volume 3, Sections 5.1.2 and 6.4).

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 (Volume 1, Section 2.2.3.2). Performance assessment sensitivity studies indicate that this factor is of low importance to performance (Volume 3, Sections 5.1.3 and 6.4). However, the judgment of the performance assessment analysts is that the influence of the factor "percolation to depth" should be similar to that of precipitation and infiltration. The latter two factors provide the input values to the percolation model; however, the amount and redistribution of water as it travels through the unsaturated zone to the drift are contained in the model for the percolation flux. A defensible percolation model is needed for calculations of water seepage into the drifts, and as shown in figures in Section 5.1.4 of Volume 3, the uncertainty in the peak dose rates depends strongly on seepage. Therefore, uncertainty in the percolation model is judged to be moderately important to calculations of the peak dose rate in TSPA.

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 (Volume 1, Section 2.2.3.2). The variability of this factor is greater than that for the average percolation flux at the repository horizon because seepage is sensitive to the variable geometry and hydrologic properties of fractures near the drift wall. Fracture flow processes can divert percolating water through the rock around the drift, bypassing the drift openings. The flow properties that control this behavior may be changed by the repository, for example, by excavation of the emplacement drifts or as a result of thermomechanical stresses. Because of the natural variability in fracture characteristics, and uncertainty about repository-induced changes, sensitivity studies show that a wide range of seepage behavior is possible given the current understanding. Because of the sensitivity of postclosure performance to the magnitude of seepage, this factor is highly important to postclosure performance (Volume 3, Sections 5.1.4 and 6.4).

Effects of Heat and Excavation on the Flow. Heat will be generated in the repository from radioactive decay of the waste, increasing temperatures in the emplacement drifts and out in the host rock for a few thousand years, until the heat output declines and the repository 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. Excavation of the drifts, mechanical loading of the rock mass by thermal stress, and eventual failure of the drift openings (i.e., rockfall) could modify the flow properties as well (Volume 1, Section 2.2.6).

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 influence the ambient flow in the host rock after the thermal period. Studies to date have not determined how thermally induced changes in the host rock will alter hydrologic properties. Some analyses have shown that changing the properties could lead to significant changes in the duration of the dry period, which could potentially change the dose curves significantly. In addition, hydrologic properties of the rock mass are expected to change because of mechanical and possibly chemical processes. Permeability changes could occur because of fracture deformation under thermal stress or because of precipitated minerals. Sorptive properties may change along potential pathways for radionuclide transport after cooldown.

When all these considerations are included in an assessment of the significance of the uncertainty in the TSPA, the influence of uncertainty in this factor is judged to be moderate. The *Performance Assessment Peer Review Panel Report* (Whipple et al. 1998) indicates that this factor merits further consideration in light of the preliminary analyses.

Dripping of Water onto the 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 reflects only the volume of water falling on the waste package, not the location of the seeps, which is contained in the seepage into drifts factor.

The specific analyses for this factor, as shown in Volume 3, Sections 5.4.3 and 6.4, show a low sensitivity of uncertainty in the TSPA. However, the dripping model used in TSPA-VA is based on several assumptions which are clearly conservative and could probably be refined to better represent performance; for example, the requirement that all the seepage entering the drift above the springline adjacent to a waste package drips onto the package. The possibility of refining this model with a modest effort is not fully captured in the quantitative sensitivity analysis. Accordingly, uncertainty in the dripping model is judged to be moderately important to calculations of the peak dose rate in TSPA.

Humidity and Temperature at the 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. Nevertheless, performance assessment sensitivity studies indicate that uncertainty in this factor is of low importance to performance (Volume 3, Sections 5.2.1 and 6.4). For the conceptual model used to represent waste package inner barrier corrosion, the rate of corrosion is not strongly affected by the presence of moisture (which may be facilitated by salts) during part of the repository thermal evolution. A different conceptual model is used to evaluate the effects of dripping water.

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 and chloride content). Performance assessment sensitivity studies indicate that this factor is highly important to the calculated peak dose rates (Volume 3, Sections 5.3.2, 5.4, and 6.4). 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 that a more realistic representation of these effects is possible, it is judged that the importance of the chemistry to postclosure performance is moderate.

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 being provided by the corrosion-resistant inner barrier. However, the outer barrier does play a role, providing structural integrity to the package. Thus the effects of rockfall or other mechanical degradation modes are expected to be mitigated by this barrier. The outer layer also provides additional protection from conditions that might accelerate the corrosion of the inner waste package layer. The performance contributed by the outer layer was not explicitly assessed by the TSPA-VA; however, when consideration is given to the potential of this barrier to increase waste package lifetime, particularly in the first 10,000 years of repository performance, the uncertainty as to the lifetime of the outer carbon steel barrier is judged to be moderately important to TSPA (Volume 3, Section 6.4).

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 judgment is that the performance up to one million years is highly sensitive to the lifetime of the corrosion-resistant inner barrier (Volume 3, Sections 5.4 and 6.4).

Some waste packages will contain barriers in addition to those identified here. For example, the high level radioactive 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.

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. The sensitivity analyses performed for the parameters used to represent this factor indicate that the relative sensitivity of its contribution to performance is low (Volume 3, Sections 5.5 and 6.4). However, there is currently no experimental or observational information upon which to base the assumptions about how much water enters the waste package. Thus, a variation in these parameters could lead to either significantly higher or lower doses. In addition, for similar reasons that the other characteristics associated with flow system are moderately important to postclosure performance, the judgment of the performance assessment analysts is that a more realistic representation of seepage into the waste package could lead to it being moderately important to the performance of the system.

Integrity of Spent Nuclear Fuel Cladding. Most of the spent nuclear 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 highly important to performance (Volume 3, Sections 5.5.1 and 6.4) for those waste packages containing clad spent nuclear fuel. On this basis the judgment is that this factor is highly important to postclosure performance.

Dissolution of Spent Nuclear 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, the 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 has not been included in the TSPA-VA. This factor, in addition to the sensitivity analyses performed for this factor, indicate that it is of moderate importance to the analysis of repository performance (Volume 3, Sections 5.5.3 and 6.4).

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 occur in later times, this factor is considered to be moderately important to postclosure performance of the system (Volume 3, Sections 5.5.4 and 6.4).

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 for certain radionuclides when compared with transport of dissolved species, which is limited by solubility and sorption processes. Colloidal transport is included in TSPA with substantial uncertainty on key parameters, and with the potential to significantly influence the peak dose rate. The performance assessment sensitivity studies indicate that formation and stability of radionuclide-bearing colloids is moderately important to postclosure performance (Volume 3, Sections 5.5.5 and 6.4).

Transport Through and Out of the Engineered Barrier System. Transport mechanisms in the waste package could affect the rate mobilized radionuclides would be released from breached waste packages. The sensitivity analyses using the current TSPA analyses indicate that this factor is of low importance to performance of the system (Volume 3, Sections 5.5.6 and 6.4). 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 waste package openings 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. Therefore, these additional considerations have led the analysts to revise the assessment of sensitivity to moderate.

Transport in 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 of the rock matrix.

The range of sensitivity analyses conducted to evaluate the significance of unsaturated zone radionuclide transport to the overall performance of the repository system are presented in Section 5.6 of Volume 3. These analyses indicate that there is less than a factor of five 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, retardations, and colloid stability) have a low importance to dose rate.

The flow and transport model used to evaluate sensitivity has inherent dispersion characteristics associated with the numerical representation of the repository horizon as a large source of radionuclides. The magnitude of dilution assumed in this representation is at least a factor of 10, which if multiplied with the sensitivity on key transport characteristics gives an overall sensitivity approaching a factor of 50. Considering all of these uncertainties, the significance of uncertainty is judged to be high.

Flow and Transport in the Saturated Zone. Flow and transport in the saturated zone is a principal factor because it provides the pathway for dissolved and colloidal radionuclides to move away from the repository and because there are widely recognized issues related to saturated zone performance. The properties (e.g., flux, effective porosity, and hydraulic structure) determine how rapidly the radionuclides can travel away from the repository. Chemical retardation processes, such as sorption and precipitation, and physical and chemical characteristics that control transport by radionuclide-bearing colloids can decrease radionuclide concentrations for the first few thousand years after radionuclides arrive at the water table and delay the peak dose rate associated with downgradient use of the water. In addition, concentrations can be reduced through dilution caused by dispersive mixing with uncontaminated water along transport pathways. Sensitivity studies using the model for saturated zone transport in TSPA-VA indicate that the significance of uncertainty for this factor is moderate (Volume 3, Sections 5.7 and 6.4). Refinement of this model is planned for the site recommendation and the LA. The significance of uncertainty could increase as the model more realistically accounts for processes that reduce radionuclide concentrations.

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. The sensitivity studies indicate that, 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 (Volume 3, Sections 5.8.2 and 6.4). This ranking of significance is primarily based on 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. This analysis assumes that the critical exposed population is the population that will be using water from the alluvial aquifer several thousands and tens of thousands of years from now.

It is likely that the environmental standard being developed by EPA and the regulatory criteria being developed by NRC will specify the location and characteristics of the populations and individuals to be assessed in the dose calculations. Once the regulations that are ultimately implemented for the Yucca Mountain repository system are defined, it is possible that the significance of the potential dilution from pumping could range from very important to unimportant from a system performance perspective. This importance would 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 judgment is that this factor is considered to have a moderate significance on post closure performance.

Biosphere Transport and Uptake. Simple models for uptake of radionuclides in which concentrations of radionuclides in water are multiplied by drinking water dose conversion factors were used in previous performance assessments. More sophisticated models of the uptake have been developed involving use of contaminated water for bathing or irrigation, uptake by plants and grazing animals, and aspects of the biosphere that could affect the concentration. These more sophisticated models used in recent performance assessments indicate the potential for either an increase in or reduction in the effect of the radionuclides. The performance assessment sensitivity studies indicate that this factor could be moderately important to the estimates (Volume 3, Sections 5.8.1 and 6.4). However, these sensitivity analyses are in part predicated on the possibility that the definition of the "average" individual of the critical group for whom the dose is being calculated changes from a "rural residential" individual to a "subsistence farmer." Given that the groups and individuals that will ultimately be considered in the consequence assessment have yet to be defined by EPA and NRC, the biosphere transport factor in the analyses, although uncertain and having an impact on performance, is considered to be of low significance.

2.2.4 Prioritization of Principal Factors

This section describes the results of a systematic approach to prioritization of technical work to address postclosure performance. The results are conditional because they are based on the VA reference design and DOE is still considering design options and alternative designs. As the design evolves, so will the prioritization of technical work.

The objective for this prioritization is to identify technical work that has the best potential to contribute to reduction of uncertainty in the postclosure safety case in a manner that is most sensitive to peak dose rate. This work has received priority in the allocation of funding and other resources.

2.2.4.1 Definitions and Discussion of Measures Used in Prioritization

The results of the are summarized in Table 2-2. The attributes of the repository safety strategy (first column) and the principal factors (third column) have been discussed previously. The remaining columns are discussed next.

TSPA Model Components. The entries in this column of Table 2-2 are the major process models that together compose the overall TSPA model (Volume 3, Section 3).

Significance of Uncertainty to TSPA. Estimates for this measure were made by considering quantitatively the effects of uncertainties associated with each principal factor on the peak dose rate calculated by TSPA. Judgments were then made taking into account limitations of the quantitative approach.

Current Confidence. The assessment of current confidence provides an indication of the degree to which the current representation of each principal factor in the VA provides an appropriate representation for the postclosure safety case that will be prepared to support the decisions for site recommendation and licensing. Current confidence is defined as an expression of the degree of certainty that the current representation of the principal factor in the VA is realistic and captures the entire range of conditions important to performance. There are two elements to the definition: "realistic" refers to how well the current representation portrays the physical processes believed to be present, and "captures the entire range of conditions" refers to how well the representation of uncertainties spans the range of potential variations in the principal factor. A seven-state scale was adopted for evaluating current confidence (one through seven, with increasing numbers indicating higher confidence). The assessment considered the VA reference design. Some ratings of current confidence may increase or decrease, if the repository design changes, because of the design alternatives evaluation (described in Section 2.4.3).

Using this definition, the current representation in TSPA of a principal factor that is unrealistically

Table 2-2. Results from Conditional Prioritization of Information Needed to Address the Principal Factors of Postclosure Performance (modified from YMP 1998)

Attributes of the Repository Safety Strategy	TSPA Model Components	Principal Factors	Significance of Uncertainties to TSPA[1]	Current Confidence[2]	Confidence Goal[3]	Priority for SR/LA[4]
Limited water contacting waste packages	Unsaturated Zone Flow	Precipitation and infiltration into the mountain	M	4	5	1
		Percolation to depth	M	3	5	2
		Seepage into drifts	H	2	5	3
	Thermal Hydrology	Effects of heat and excavation on flow:				
		A. Mountain scale	M_b	1	2	1
		B. Drift scale	M_b	2	4	2
		Dripping onto waste packages	M	2	4	2
		Humidity and temperature at waste packages	$L_{b,c}$	5	4	0[5]
Long waste package lifetime	Near-Field Geochemical Environment	Chemistry of water on waste packages	M	3	5	2
	Waste Package Degradation	Integrity of outer carbon steel waste package barrier	M_a	4	5	1
		Integrity of inner corrosion-resistant waste package barrier	$H_{a,b}$	3	6	3
		Seepage into waste packages	M	3	3	0
Low rate of release of radio-nuclides from breached waste packages	Waste Form Alteration and Mobilization	Integrity of spent nuclear fuel cladding	H_a	3	5	2
		Dissolution of spent nuclear fuel and glass waste forms	$M_{b,c}$	4	5	1
		Neptunium solubility	$M_{b,c}$	4	5	1
		Formation and transport of radionu-clide-bearing colloids	$M_{b,c}$	2	4	2
		Transport through and out of the engineered barrier system	$M_{b,c}$	3	4	1
Radionuclide concentration reduction during transport from the waste packages	Unsaturated Zone Transport	Transport through the unsaturated zone	H_a	2	5	3
	Saturated Zone Flow and Transport	Flow and transport in the saturated zone	M	2	3	1
		Dilution from pumping	M	5	5	0
	Biosphere Transport	Biosphere transport and uptake	L	5	5	0

1 Significance of uncertainties to TSPA is specified as low (L), moderate (M), or high (H), where these terms mean: L (low) if possible variations in the factor, including absence of the factor, do not change system performance estimates by more than a factor of 5; M (moderate) if variations in the factor could change the system performance estimates by more than a factor of 5 but less than a factor of 50; and H (high) if variations could change the estimates by more than a factor of 50. These factors are approximate because of the nature of the approximations used in the analysis. Sensitivity analyses of the principal factors are divided into three periods: (a) significance to 10,000 years, (b) significance from 10,000 to 100,000 years, and (c) significance from 100,000 to 1,000,000 years (see Volume 3, Section 6.4). Lack of time period designators (a, b, or c) indicates that the principal factor has the same uncertainty throughout all time periods; otherwise, the period designators indicate periods of uncertainty.

2 Current confidence is an expression of the degree of certainty that the current representation (of the principal factor in TSPA) is realistic and captures the entire range of conditions important to performance. The confidence scale is one through seven, with seven indicating higher confidence.

3 The confidence goal is the confidence level that is both feasible and desirable for the site recommendation (SR) and the LA. The confidence scale is one through seven, with seven indicating higher confidence.

4 Priority for the SR/LA is the difference between the confidence goal and the current confidence, rounded to the nearest integer.

5 The calculated number of −1 for this principal factor has the same meaning as a zero.

narrow in its consideration of uncertainties, meaning that alternative and potentially viable models or parameter values lie outside those considered in the VA, would be given a relatively low level of current confidence. Likewise, a representation that includes an unrealistically large range of uncertainties or a range that is skewed to one side of a realistic range, would be assigned relatively low current confidence. Highest current confidence is assigned to those principal factors for which a high degree of certainty exists that the current representation is realistic and reasonably spans the range of possible uncertainties.

Confidence Goal. This assessment is defined as the confidence level that is both feasible and desirable for a site suitability determination for site recommendation and preparation of a defensible LA. This assessment is a judgment that is based on input from both principal investigators and technical managers. Here, "feasible" means the work can be accomplished in time to provide input to the Secretary's site recommendation decision, planned for 2001, and, if the site is recommended and designated, to be incorporated in a LA submittal in 2002. "Desirable" is defined as being significant to TSPA and important to the defensibility of the technical basis for the site recommendation decision and licensing submittals. As in the assessment of current confidence, a scale of one through seven was adopted for this assessment, with seven indicating the highest confidence.

The confidence goals for the principal factors in Table 2-2 are used to establish priorities for work that DOE has planned in support of the site recommendation decision and LA, as discussed next. However, it should be noted that focussed site investigations, design studies, and performance assessments will continue after the site recommendation decision and, if the site is recommended and designated, after submittal of the LA. Some of this work is planned long-term testing and monitoring that is part of the performance confirmation program (see Sections 2.2.1.5, 3.1.6, and 3.2.1.7). Also, additional information-gathering activities may be required to respond to questions that arise in the course of the licensing review. By the time of construction authorization, these post-LA studies should further increase the confidence levels for some of the principal factors. During each subsequent phase—construction, operation, monitoring, and closing—performance confirmation should continue to increase confidence in scientific understanding.

Priority for Site Recommendation and License Application Decisions. This column of Table 2-2 expresses the relative priority of each principal factor, and the information that will be obtained to address that factor, in preparation for the site recommendation and LA decisions. The rating value is the difference between the confidence goal and current confidence for each principal factor, expressed as the numerical difference between the two. A small value indicates that current confidence is already comparable to the confidence level that is desirable and can be achieved for the site recommendation and the LA. A larger value indicates a higher priority for gaining confidence before the site recommendation and LA decisions.

In the case of one principal factor (humidity and temperature at the waste package), the confidence goal is less than the current confidence. In this case, the significance to TSPA was judged to be lower, because of TSPA-VA insights, than it was previously judged to be.

The ratings of priority for the site recommendation and the LA in Table 2-2 give priorities for information needs but not for technical work, because the information needs are not expressions of work scope. Other considerations such as cost and programmatic needs enter into assessments of priorities for future technical work (technical work plans are presented in Section 3). Rather, the priority assigned to each factor is a useful representation of the gain of knowledge that is desirable and reasonably achievable for increasing confidence for the site recommendation and LA decisions.

The following summary of the prioritization of information needed to address each principal factor briefly describes the technical work that DOE has identified as being both desirable and feasible.

2.2.4.2 Assessments for Prioritization of Principal Factors

Precipitation and Infiltration into the Mountain. *Current Representation in TSPA*–The current representation of this principal factor consists of the spatial distribution of net infiltration and the time history (or temporal variation) of infiltration as a function of time over one million years. The spatial distribution, which comes from an analysis of net infiltration under present-day conditions, accounts for variations in such characteristics as precipitation, topography, rock type, and vegetation (Volume 1, Section 2.2.3.2). The model is calibrated from moisture profiles in boreholes and from precipitation records. The resulting net infiltration map for present conditions shows this spatial distribution at a grid spacing of about 30 m (98 ft).

To express the uncertainty in the assessment of net infiltration under present conditions, a variation from 3X to 1/3 X is used, where X is the estimated value of expected present infiltration at a point. This variation is consistent with the range of conditions inferred from analysis of thermal gradients measured in seven boreholes. The probabilities of occurrence that are associated with the variations are: 3X has 10 percent probability, 1X has 60 percent probability, and 1/3 X has 30 percent probability. Thus the expected values of infiltration have the greatest likelihood. These factors are applied uniformly to all of the grid points on the net infiltration map. It is assumed in the present TSPA model for the VA that the 3X to 1/3 X uncertainty in net infiltration applies not only to the present climate regime, but also to what are defined as the long-term average and the superpluvial regimes. The long-term average regime is assumed to prevail throughout most of the next million years. The superpluvial regime is used to represent interludes during this period when climate is wetter and more variable than the long-term average.

The net infiltration estimates for long-term average and superpluvial are assessed based on paleoclimate studies (e.g., oxygen isotope studies on cores, pack-rat middens, and vegetation changes) and analogs, with the assumption that the past can be used as a predictor of the future (Volume 1, Section 2.2.3.2; Volume 3, Section 3.1.1.2). The modeled climate changes among three states: dry or present-day climate; long-term average climate, which is about twice the dry-climate precipitation; and superpluvial climate, which is about three times the dry-climate precipitation. The length of the first dry climate, or current climate, is sampled uniformly between 0 and 10,000 years, with an expected value of 5,000 years. The length of subsequent dry climates is sampled uniformly between 0 and 20,000 years, with an average of 10,000 years. The duration of long-term average climates is sampled uniformly between 80,000 and 100,000 years. Dry and long-term average climate alternate from the time of repository closure until the long-term average climate spans the 250,000-year mark. The last part of that long-term average period is replaced by a superpluvial climate of sampled duration. Then the climate model returns to alternating dry and long-term average climates until the long-term average climate that spans the time of 400,000 years after the first superpluvial. As mentioned previously, uncertainty of a factor of 3X to 1/3 X is also included in the magnitude of the net infiltration during these periods.

Significance of Uncertainties to TSPA–It is difficult to discern from the TSPA sensitivity analyses (Volume 3, Section 5.1.2) the magnitude of the variation in peak dose rate that results from the variation in net infiltration. There is some evidence that the impact on peak dose rate is greater than a factor of 10 (thus suggesting a "moderate" significance), but other plots show less sensitivity. The assessment of moderate significance is adopted for this prioritization.

Current Confidence–The current representation of net infiltration in the TSPA is assigned a current confidence of "4." The present model assumes the same relative spatial distribution for three climate states (i.e., the ambient net infiltration map is used for all periods, multiplied by appropriate factors to account for a difference in average infiltration during wetter periods). There is some indication that, in fact, during wetter periods the spatial distribution may change (e.g., channels and gullies during wet periods may be areas of very high infiltration that are not presently represented in these maps). Also, the magnitude and duration of future climate states appears to be highly uncertain. There is some evidence that suggests the duration of dry periods is more likely to be 20,000 years than the 10,000-year expected value used in the TSPA. The analysis of net infiltration suggest that the possibility of a 3X uncertainty in net infiltration may provide unrealistically wide bounds to the infiltration (Volume 3, Section 5.1.2), although there are uncertainties in the techniques used to estimate net infiltration. The "step function" representation of the duration of change between climate intervals (Volume 3, Section 3.1.1.1) is probably unrealistic, and it is difficult to judge the degree of conservatism.

Confidence Goal–The assessment of confidence goal takes into account the desired confidence, both because of importance to TSPA and from the standpoint of defensibility, as well as whether that confidence can reasonably be achieved for site recommendation and the LA. It was concluded by the team that a confidence goal of "5" was appropriate. There are few new measurements that can be conducted in this time period. However, modeling and literature analyses can be accomplished to increase confidence in the potential spatial distribution of net infiltration, especially during wetter climates, and to evaluate the potential uncertainties in the magnitude of net infiltration during the wetter periods, especially the long-term average. These analyses could include analogs to Yucca Mountain, particularly those having climates comparable to those expected during wetter periods, to look at spatial variability in net infiltration for these analogs and whether the step changes are appropriate. The significance to TSPA of this principal factor is moderate; the above-mentioned activities will provide the needed defensibility.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence, "4," and the confidence goal, "5," the numerical difference between these two assessments (5–4 = 1) provides the priority for site recommendation and LA of "1" for this principal factor.

Percolation to Depth. *Current Representation in TSPA*–The current representation of percolation flux at the repository horizon is derived from estimates of the spatial distribution of net infiltration, field and laboratory data and is calibrated against a variety of data sets (Volume 1, Section 2.2.3.2). Estimates of infiltration depicted at a 30-m (98-ft) grid spacing are generalized to a coarser grid of 100-m to 200-m (328-ft to 656-ft) block size. Using the base case hydraulic parameter property sets for both the rock matrix and the fractures, the spatial distribution of flux is assessed over the repository region in the site-scale unsaturated zone flow model. The hydraulic properties used for the partially saturated fractures are obtained from inverse modeling of an equivalent continuum representation obtained from the calibrated site-scale model (Volume 3, Section 3.1.1.3). The site-scale model is calibrated against chlorine-36 data, temperature gradients, porewater chloride mass balance, carbon-14 data, matrix saturation, and observations of perched water (Volume 1, Section 2.2.3.2). The repository layout is then divided into six regions and the spatial variability of the matrix and fracture components of flow at the repository horizon are represented by probability density functions for each region. The uncertainties in the matrix and fracture property sets are incorporated into the probability density functions, for both the matrix and fracture components, for each region.

Even with the numbers of data used to calibrate the percolation model, the data still permit a range of uncertainty. The current model includes 39 different representations that are believed to capture the range of these conceptual models. The TSPA analyses currently consider each of these 39 flow fields as inputs.

Significance of Uncertainties to TSPA–Sensitivity analysis of the variation in percolation flux shows that the peak dose rate varies by a factor of approximately 5 to 10 or less during all three periods of interest (Volume 3, Section 5.1.3). However, percolation is judged to be equally important as infiltration. Accordingly, an assessment of moderate significance is used for this prioritization.

Current Confidence–The assessment of the current confidence in the representation of percolation flux is "3." The current representation is believed to incorporate many uncertainties, but may not span the range of reasonable possibilities. Data on hydraulic property sets and net infiltration are incorporated. However, because of the rather coarse spatial discretization it averages out local variations in flux. Finer-scale features such as faults and major fractures may be important when the model is used as input to the seepage model. In addition, although the probability density functions for each of the six regions appear to capture the average spatial variability in flux within the region it is not clear how they incorporate the uncertainty in flux at any given location, other than through the uncertainties in the property sets. Debate exists about the effective porosities being used in the model, with some evidence supporting higher effective porosities than currently used. The limited number of flow fields also contributes to the lack of certainty in whether a full range of behaviors is represented.

Confidence Goal–The assessed confidence goal for percolation flux is "5," which accounts for improvements that can readily be made in the representation for the site recommendation and the LA. Data from the ongoing niche tests and the cross drift that is presently under construction will be useful in calibrating the present model, as well as for further defining the uncertainties. Ongoing efforts to further refine the grid scale of the model to a 30-m (98-ft) grid block, as well as to include local faults, will help to produce a more realistic representation of the spatial distribution of flux on the drift scale. The defensibility of this principal factor will be important for the site recommendation and the LA.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "3," and the confidence goal "5," the numerical difference between these two assessments (5–3 = 2) provides a priority for site recommendation and the LA of "2" for this principal factor.

Seepage into Drifts. *Current Representation in TSPA*–In situ measurements have provided useful information about seepage, and modeling studies calibrated to these measurements have provided indications of the fraction of percolation that would actually contribute to seepage under different conditions. Estimates of 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. Current information on drift seepage is summarized in Section 2.2.3.2 of Volume 1 and Section 3.1.1.4 of Volume 3.

Using the percolation flux distribution as input (i.e., the fracture and matrix flux distributions for each of the six regions), seepage into the drifts is simulated using three-dimensional single-continuum models that include a spatially variable permeability field to represent the fracture network (Volume 1, Section 2.2.3.2). The outputs are the seepage fraction as a function of percolation flux and the seepage flow rate as a function of percolation flux. The location of seepage is then simulated from a stochastic representation of hydrologic properties to simulate the spatial distribution of seepage on the drift wall. The grid block size of the seepage model is 0.5 m (2 ft) at the drift wall, and increases to several meters at greater distances from the drifts. At present, the seepage model is approximately calibrated against limited data developed from the niche tests in the Exploratory Studies Facility.

Significance of Uncertainties to TSPA–Sensitivity tests show that variations in seepage into the drifts can lead to variations in peak dose rate of more than a factor of 50. Therefore, the assessment of high significance of uncertainty was accepted for this prioritization.

Current Confidence–The assessment of the current confidence in the seepage into the drifts is "2." There is relatively low confidence that the representation spans the range of conditions affecting seepage. There are large uncertainties in characterizing seepage within a heterogeneous system, particularly at the local scale required by the model. Uncertainties in rock and fracture properties are significant at these scales, and issues related to the drift geometry may be important but are not presently accounted for. The shape of the drift is assumed to be circular, and the model presently does not account for irregular shape caused by drift collapse, the presence of a drift lining, or small-scale variations in drift geometry such as roughness of the drift wall. Despite having a realistic physical basis, the present seepage model is based upon, and calibrated against, very few measured data.

Confidence Goal–Field tests of seepage in niches along the Exploratory Studies Facility main drift and cross drift are systematically measuring seepage into drifts, as a function of different water release rates above the niche openings, for all three tuff units in the repository level. Seepage thresholds, or average percolation flux values below which dripping onto waste packages is unlikely to occur, will be determined for the different tuff units, fractured zones, two sites near a fault, and one site below a tuff interface. In addition, with the cross drift passing directly over the Exploratory Studies Facility main drift, the region around the crossover point will provide an opportunity to conduct large-scale seepage measurements. Seepage into the openings below will be induced by injection from the openings above. Results from all of the experiments will be available to support the site recommendation and LA decisions.

The confidence goal for seepage is judged to be "5." The high importance of this issue to TSPA, as well as the strong need for its defensibility, mean that it is desirable to gain a considerably higher level of confidence than currently exists. However, the capability of doing so for the site recommendation and the LA tempers the assessment of confidence goal. It is believed that the ongoing niche studies will provide very valuable information on which to refine the seepage model, and planned modeling studies will provide better expression and incorporation of uncertainties (e.g., incorporation of drift geometry issues, refined portrayal of hydrologic units, better definition of seep locations).

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "2," and the confidence goal "5," the numerical difference between these two assessments (5–2 = 3) provides a priority for site recommendation and the LA of "3" for this principal factor. This relatively high priority is caused largely by the high significance of the principal factor to TSPA, the need for defensibility, and the relatively low present level of confidence in the current representation.

Effects of Heat and Excavation on Flow. *Current Representation in TSPA*–The present modeling of the thermal effects on flow is divided into a consideration of mountain-scale effects and drift-scale effects. Most of the hydrologic effects of the thermal pulse occur during the first few thousand years, and the temperature in the repository is calculated to return to nearly ambient by 10,000 years (Volume 1, Section 2.2.6.2, and Volume 3, Section 3.2.2). The mountain-scale model is a generalized equivalent continuum model that uses the hydraulic property sets developed for the unsaturated zone flow model. The drift-scale model is a dual permeability model that represents the fracture and matrix continua separately and characterizes the fracture-matrix interaction. The grid size of the mountain-scale model is 100 m (328 ft), and for the drift-scale model it is 0.5 m (2 ft). The mountain-scale model uses a smeared heat source, and repository edge effects are predicted where heat is dissipated into the unheated rock mass adjacent to the repository. The drift-scale model represents heat production within the waste packages, and accounts for the heat generation of different types and configurations of waste packages. The outputs of the drift-scale model are the temperature, relative humidity, and air mass-fraction, which are used in the TSPA models that pertain to the engineered barriers. The outputs of the mountain-scale model are flux and air mass-fraction, which are used to constrain the large-scale temperature field for the drift-scale model.

The present TSPA model assumes that the hydrologic properties of the host rock are the same before, during, and after the thermal period. Limited modeling suggests that thermal, hydrologic, and mechanical effects on hydraulic properties can be ignored at both the mountain and drift scales. That is, while it is acknowledged that fracture permeabilities may change during the thermal period (e.g., fractures may close or open because of compressive thermal stresses, thus changing permeabilities), the magnitudes of these changes probably fall within the range of natural variability under ambient conditions. For example, the single heater test showed changes in measured fracture permeability of less than a factor of 1/2 to 1/5. This is within the range of bulk permeability measured in situ, which typically spans more than three orders of magnitude. Although permeability changes caused by heat are being ignored in the TSPA for now, the importance of this potential effect is recognized.

Significance of Uncertainties to TSPA–It is difficult to evaluate the significance that this principal factor has on performance, and additional documentation is needed. An "alternative" property set that is believed to be more representative of the non-lithophysal zone in the southeast part of the repository leads to a decrease in peak dose rate of slightly greater than 10X, thus suggesting a "moderate" significance to uncertainty for drift-scale effects. Another way of assessing the significance is by comparison to significance of the seepage model. The variation in the property sets at the location of seeps is comparable to the expected change in properties caused by thermal influences. These changes can be shown to result in a 10X to 30X change in peak dose rate, thus the assessment of "moderate" significance of uncertainty is accepted for this prioritization (Volume 3, Section 5.2.1). By analogy to the drift-scale result, "moderate" significance of uncertainty is also accepted for mountain-scale effects.

Current Confidence. The assessments of current confidence in mountain-scale effects of heat and excavation and drift-scale effects are given separately.

Current Confidence: Mountain-Scale Effects of Heat and Excavation on Flow–The current confidence in the mountain-scale representation is judged to be "1." This indicates low certainty that the current representation spans the range of possibilities realistically. Key uncertainties in both the mountain-scale and drift-scale models are the rates at which rewetting occurs as the heat generation decreases, and the attendant changes in moisture and hydraulic properties. Also, the laboratory data that provide hydraulic property sets are developed primarily from the middle non-lithophysal member of the Topopah Spring formation (which comprises only about 25 percent of the repository horizon) and far fewer data on the lower lithophysal zone (which comprises about 65 percent of the repository horizon). Possible chemical changes such as decreases in fracture or matrix permeability caused by mineral precipitation, alteration of mineral assemblages, or development of a mineral "cap" have not been included in the current representation.

Other reasons for the relatively low confidence are that there are very few data available (with the exception of the single heater test and the large block test) on which to base the representations and nearly all of the results come from predictive models. Potentially important processes are left out of the current models, thus there is considerable uncertainty about whether the current models span the range of possibilities.

Current Confidence: Drift-Scale Effects of Heat and Excavation on Flow–Current confidence in the drift-scale representation is judged to be "2." The uncertainties presently represented likely do not span the range of possibilities, such as alternate property sets and alternate models of rewetting phenomena. An uncertainty in the drift-scale model is the assumption that the rock properties, including those governing rewetting, are not affected by the thermal pulse. Property sets have not been developed for very high flux, such as might develop during condensation and refluxing. Likewise, the potential changes in fracture-matrix interaction caused by, for example, precipitation or dissolution of minerals lining fracture walls, have not been included in the present model.

Mechanical stability of drifts during the thermal period has been considered only in sensitivity studies and is not part of the base case. It is assumed in the sensitivity studies that total collapse of all drifts occurs at 1,000 years and the thermal (i.e., change in heat transfer mechanisms) and hydrologic properties (e.g., porosity) of the debris that fall into the drift have been included in the hydrologic models. There has not been a specific consideration of the stability of the drifts through time as a function of thermal loading.

Confidence Goal. Mountain-scale and drift-scale effects of heat and excavation are assessed separately.

Confidence Goal: Mountain-Scale Effects of Heat and Excavation on Flow–The confidence goal for the mountain-scale representation is judged to be "2." It is generally concluded that with the possible exception of refinements in modeling, there are limited opportunities to improve confidence before submittal of the site recommendation and the LA. Using discrete heat sources to represent emplacement drifts and using a dual-permeability model for the rock could help build confidence. The desired increase in confidence is low because it is judged that high defensibility of this model likely will not be needed at submittal of the site recommendation and the LA. It is not clear how significant this principal factor is to TSPA.

Confidence Goal: Drift-Scale Effects of Heat and Excavation on Flow–The confidence goal for the drift-scale representation is judged to be "4." There are ongoing and planned activities that will increase confidence in this factor. The interim results of the drift-scale test may serve to better constrain the models and to yield more appropriate property sets for the thermally perturbed condition. Additional drainage and rewetting experiments in additional units may be possible. Because of the importance of this issue to several review groups, it is judged that this principal factor must be addressed in order to increase defensibility at the submittal of the site recommendation and the LA. The capability to do this for the site recommendation and the LA places limits on how much confidence in this factor can be improved.

Priority for Site Recommendation and LA–For the mountain-scale representation and based on the assessment of the current confidence "1," and the confidence goal "2," the numerical difference between these two assessments (2–1 = 1) provides a priority for the site recommendation and LA of "1" for this principal factor. For the drift-scale representation and based on the assessment of the current confidence "2," and the confidence goal "4," the numerical difference between these two assessments (4–2 = 2) provides a priority for the site recommendation and LA of "2" for this principal factor.

Dripping onto Waste Packages. *Current Representation in TSPA*–The dripping model in the current TSPA determines which waste packages receive drips and the volume of water that drips. The dripping model is based solely on the previously discussed drift seepage model. The seepage model gives the spatial location and volume of seepage at the drift wall. It is then assumed that seeps lying above the drift spring line and adjacent to or above a waste package fall on the waste package, thus the waste package width is assumed to be the same as the drift diameter. For this implementation, variability and uncertainty in the seepage model translate directly into variability and uncertainty in the dripping model. At the scale of multiple waste packages, it is assumed that the location of drips stays constant through time given the same climatic state (i.e., the wet waste packages continue to see drips and the dry packages remain dry). With wetter climate states, the number of seeps, and thus the population of waste packages with drips, increases, but the locations of previous drips remain the same.

Significance of Uncertainties to TSPA–Sensitivity studies show that the variation in peak dose that results from alternative conceptualizations for the dripping model, is less than a factor of 10 (Volume 3, Section 5.4.3). It is judged that because the dripping model primarily influences the volume of water that drips on the waste package, uncertainty is potentially more important later in the performance period (after 10,000 years) when the waste packages begin to fail because of corrosion. After breach of a waste package, the volume of water that passes through the waste package will directly affect the release rate of radionuclides from the waste form, particularly the solubility-limited species such as neptunium. Therefore, the assessment of "moderate" significance of uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of dripping is judged to be "2." The representation of the waste package as having the width of the whole drift is conservative, and the relatively sparse modeling information that shows seeps mostly near the spring line is questionable. Also, the assumptions that all seeps above the spring line drip onto the waste package and that the drip locations are stationary, are conservative but may not be realistic. Observations from Rainier Mesa and Apache Leap suggest that drip locations vary through time.

Confidence Goal–The confidence goal for the representation of dripping is judged to be "4." It is desirable to increase the realism in the model before submittal of the site recommendation and the LA, and this can be readily accomplished. The data coming from the niche tests and the Alcove 1 tests in the Exploratory Studies Facility may provide useful information in submittal of the site recommendation and the LA. Given their moderate significance to TSPA, there is a need for increased defensibility in this principal factor before licensing.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "2," and the confidence goal "4," the numerical difference between these two assessments (4–2 = 2) provides a priority for the site recommendation and LA of "2" for this principal factor.

Humidity and Temperature at Waste Packages. *Current Representation in TSPA*–The assessments of temperature and relative humidity at the waste package surface use information from mountain-scale thermal models, and drift-scale thermal-hydrologic flow models (Volume 1, Section 2.2). The variability in the heat output between waste packages is included by using a three-dimensional thermal-hydrologic model of an idealized drift segment that incorporates the various waste package types in proportion to their expected inventories (i.e., boiling water reactor and pressurized water reactor spent nuclear fuel, high level radioactive waste glass, and co-disposal). The mountain-scale model assumes conductive heat transfer only, accounting for repository edge effects, and is linked to the drift-scale model. The models use the percolation flux distribution from the six repository regions as input. Within the drifts, radiant heat transfer is the dominant mode because the VA reference design does not include drift backfill. Although not included in the present model, rockfall was evaluated in sensitivity analyses. The presence of rockfall debris in the model causes a change to conduction and convection-dominated heat transfer, with the result that relative humidity on the waste package surface increases somewhat sooner during the cool down period.

Significance of Uncertainties to TSPA–One means of evaluating the significance to TSPA is to look at the difference in peak dose rate for two very different thermal loading configurations, 25 and 85 MTU per acre. The difference between 25 and 85 MTU per acre mass loading results in temperature and humidity changes that produce a factor of two difference in peak dose rate (Volume 3, Section 5.2). The assessment of "low" significance to uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the relative humidity and temperature on the waste package representation is judged to be "5." It is felt that the present model appears to be realistic and to span most of the range of possibilities. However, it does not incorporate some effects, such as those that might result from rockfall into the drifts.

Confidence Goal–The confidence goal in the relative humidity and temperature on the waste package representation is judged to be "4." It is felt that the current model is adequate for purposes of the site recommendation and the LA.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "5," and the confidence goal "4," the numerical difference between these two assessments (4–5 = -1) provides a priority for the site recommendation and LA of -1, which is assumed to be zero for this principal factor.

Chemistry of Water on Waste Packages. *Current Representation in TSPA*–This principal factor deals with the bulk chemistry of water that contacts the waste package surface. The potential development of localized chemistries on the waste package is dealt with in the principal factors related to the integrity of the corrosion-allowance and corrosion-resistant waste package barriers.

The assessment of bulk chemistry begins with the assumption that the chemical composition of water vapor is essentially that of distilled water. Drips have the chemistry of J-13 water and react with iron oxides on the waste package surface. Reactions with the concrete lining are not included in the TSPA base case because it is assumed that the lining will collapse after about 300 to 500 years and the debris will be lying on the floor of the drift below the elevation of the waste packages. Nevertheless, sensitivity analyses were conducted by assuming that concrete-modified waters will interact with the waste packages, and the results showed significant differences in performance because of enhanced corrosion of the carbon steel waste package outer barrier (Volume 3, Section 5.3.2). The chemistry model included time-histories of pH, total carbonate, ionic strength, and oxygen fugacity.

Significance of Uncertainties to TSPA–If water chemistry is capable of accelerating waste package corrosion, then differences in peak dose rate may be a factor of 50 or more. However, experiments with high pH water and waste package materials have shown no evidence for enhanced localized corrosion, even with water that has interacted with concrete. Sensitivity calculations indicate low sensitivity of peak dose rate to water chemistry (Volume 3, Section 5.3.2), but there may be other processes by which water chemistry could be more significant. On balance, the assessment of "moderate" significance to uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of chemistry of water on the waste package is judged to be "3." The model appears to be realistic and is consistent with the findings from the single heater test (Volume 3, Section 3.3.1.2) but other uncertainties exist that lie outside of conceptual basis for the model. These uncertainties include the potential effects of drift wall interactions, gas compositions, microbial influence, and the effects of introduced materials. The single heater test water samples represented an early stage in the evolution of water composition projected to occur in the repository host rock. The issue of concrete-modified chemistries has not yet been addressed and may introduce significant uncertainties.

Confidence Goal–The confidence goal in the representation of chemistry of water on the waste package is judged to be "5." Confidence will be improved by further thermodynamic analyses of various constituents (e.g., nitrates and sulfates), evaluation of the stability of these constituents, and incorporation of the test data on high pH waters. This principal factor is moderately significant to TSPA and it is judged that this principal factor will be important to defensibility at the time of submittal of the site recommendation and the LA.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "3," and the confidence goal "5," the numerical difference between these two assessments (5–3 = 2) provides a priority for the site recommendation and LA of "2" for this principal factor.

Integrity of Outer Carbon Steel Waste Package Barrier. *Current Representation in TSPA*–The integrity of the outer carbon steel barrier of the waste package includes consideration of juvenile failures and general corrosion, and of the possibility of failure caused by rockfall. The present corrosion model is developed from literature data as well as data developed by the project for a variety of environmental conditions (Volume 3, Section 3.4.1). The corrosion mechanism is assumed to be general corrosion and the corrosion rate model depends on temperature only. There is a relative humidity threshold in the model at which humid-air corrosion begins. The current base case model does not include the potential for accelerated corrosion caused by modified water chemistry, but this has been evaluated in sensitivity studies. The data base for the corrosion model includes metals similar to the Alloy A516 carbon steel of the VA reference design. Uncertainties in corrosion rate and the relative humidity threshold are included in the analysis.

Juvenile failures, such as those resulting from manufacturing defects, are modeled in the following way. The period of juvenile failure is considered to be from the time of emplacement to 1,000 years after emplacement. The probability distribution for juvenile failures, based upon analysis of Canadian single-walled and dual-walled canisters data, assumes a total of 1 to 10 waste package failures in this period, with an expected value of one package at 1,000 years (Volume 3, Section 3.4.1.4). In the base case, juvenile failures are assumed to be large openings the same size as patch failures (16 by 16 cm, or about 7 by 7 in.).

Significance of Uncertainties to TSPA–The significance of the uncertainty in integrity of the carbon steel outer barrier is most important during the first 10,000 years. By analogy to other factors that influence waste package integrity during this time (e.g. dripping onto waste package) the significance of uncertainty is "moderate." This assessment is accepted for this prioritization.

Current Confidence–The current confidence in the representation of the integrity of the outer barrier of the waste package is judged to be "4." Although the present corrosion model appears to be representative and realistic, there are some uncertainties that may fall outside of those presently included. It is not clear that all data used to constrain the corrosion rate model and its uncertainties represent the VA reference material. Additional uncertainties include the influence of microbiologically influenced corrosion, which has been shown in sensitivity analyses to lead to a factor of 5 increase in corrosion rate. Stress corrosion cracking and pitting modes of corrosion are not included. Rockfall has not been explicitly included, although studies of the rock size distribution of rockfalls for the seismic stability analysis suggest that the rocks are not large enough to damage the intact carbon steel barrier. Preferential weld attack has not been considered.

Confidence Goal–The confidence goal in the representation of the integrity of the outer carbon steel barrier of the waste package is judged to be "5." This assessment reflects the moderate significance to TSPA, the need for a defensible integrity model, and the capability to address these uncertainties for site recommendation and LA. Most of the uncertainties described are being addressed in the ongoing corrosion-testing program or in performance assessment modeling.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "4," and the confidence goal "5," the numerical difference between these two assessments (5–4 = 1) provides a priority for the site recommendation and LA of "1" for this principal factor.

Integrity of Inner Corrosion Resistant Waste Package Barrier. *Current Representation in TSPA*–The corrosion rate of the inner corrosion resistant material is conditional on three chemistries: the bulk environmental conditions (drips or humid air); localized, occluded chemistries such as at the interface of the outer and inner waste package barriers; and chemistries within pits and crevices (Volume 3, Section 3.4.1.3). The first two of these chemistries are included in the current representation. Under humid air conditions, a general corrosion rate model is used that is developed from a number of data points on high-nickel alloy ASTM B 575 Alloy N06022 (Alloy 22) under a wide range of environmental conditions, including some that are far more aggressive than those considered possible at the repository. In the TSPA model, the rate of localized corrosion of Alloy 22 is based only on temperature, using all the experimental data without discriminating aggressive environments, and is therefore probably quite conservative. Corrosion rates for Alloy 22 and their uncertainties are included for environmental conditions ranging from relatively benign to aggressive, expressed as acidity and oxidizing potential. The aggressive conditions are intended to simulate the conditions that might develop, such as those caused by microbiologically influenced corrosion, high chloride content, and crevices between the outer barrier and inner barrier material, even though these effects are not explicitly modeled. Considerable uncertainties exist as to whether these aggressive environmental scenarios are possible at the repository. All of the variation in the corrosion model is divided into equal components of spatial variability and uncertainty.

Juvenile failures are modeled as failures of the entire waste package; that is, the failures of the inner waste package barrier are completely correlated with the failures of the outer barrier. Accordingly, the juvenile failure distribution is the same as that described earlier; a log-uniform distribution ranging from 1 to 10 waste package failures assumed to occur by 1,000 years.

Significance of Uncertainties to TSPA–The uncertainties in the present inner barrier model lead to over two orders of magnitude variation in the peak dose rate, especially at longer periods (Volume 3, Sections 4.3.2.1 and 5.4). The assessment of high significance to uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of the integrity of the inner corrosion resistant barrier of the waste package is judged to be "3." It is not clear that the present model realistically bounds the range of possibilities, and it may be too conservative in some areas. Uncertainties not considered in the model are the potential for corrosion of the outer barrier material at the interface of the inner and outer barriers leading to compression of the inner barrier, fabrication and weld effects, galvanic coupling, and stress corrosion cracking. These effects are currently assumed to be relatively insignificant. In addition, aging and possible phase changes of Alloy 22 and hydrogen embrittlement are not currently considered. The incorporation of measured corrosion rates from tests in very aggressive conditions, unlike those expected for the repository environment, contributes to the uncertainty.

Confidence Goal–The confidence goal in the representation of the integrity of the inner corrosion resistant barrier of the waste package is judged to be "6." This relatively high rating reflects the high desirability of increasing confidence, because of the high significance to TSPA and the potential gain in defensibility at submittal of the site recommendation and the LA. In addition, the likelihood of increasing confidence is judged to be relatively high. Corrosion tests currently underway will in the short term provide additional information on the initiation and repassivation of Alloy 22, which can be used in the TSPA model. Model improvements can be made to better define the ranges of localized chemical environments that need to be considered.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "3," and the confidence goal "6," the numerical difference between these two assessments (6–3 = 3) provides a priority for the site recommendation and LA of "3" for this principal factor.

Seepage into Waste Packages. *Current Representation in TSPA*–The dripping model (discussed previously) determines the location and volume of water that will drip on the waste packages. The corrosion models for the inner and outer waste package barriers dictate the time-history of breaches of the waste packages and the size of the openings that will develop (e.g., pit and patch failures). The current model for seepage into waste packages assumes that if any part of a waste package is contacted by drips, the entire waste package surface is wetted (Volume 3, Section 3.4.2). The amount of water that enters a waste package is equal to the fraction of the open area on the waste package (caused by pits and larger openings) times the volume of water dripping on the waste package. No flow on the waste package is assumed and there is no "capture" area around a particular opening assumed. However, there is an assumed uncertainty in the fraction captured of a factor of 1 to 10 (with a uniform distribution assumed), which tends to account for capture area larger than opening size.

Significance of Uncertainties to TSPA–Sensitivity tests show that during the first 10,000 years the total area of openings in partially failed waste packages is limited, and the volume of water entering the packages is relatively small (Volume 3, Section 5.5.1). During this period the release of solubility-limited species such as neptunium is limited, while soluble species such as technetium are readily released, so the resulting dose rate is insensitive to the rate of water entry. At later times when the volumes are higher, radionuclides such as neptunium will be mobilized in direct proportion to the flow rate of water. Accordingly, the assessment of "moderate" significance of uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of seepage into the waste package is judged to be "3." Although the model is judged to be relatively simple, it is unclear if it captures the range of processes that might occur as water enters the waste package. There are mechanisms such as flow on and off the waste package that are not captured in the present model. Water enters the waste package without any diffusive delay through corrosion products that may block the opening.

Confidence Goal–The confidence goal in the representation of seepage into the waste package is judged to be "3." This reflects both the relatively low desirability of increasing confidence from significance to TSPA and defensibility points of view. It is difficult to envision what types of studies can be done to increase confidence in this factor other than more realistic modeling of the physical processes with a basis in experimental work that should not be started until the LA reference design has been selected.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "3," and the confidence goal "3," the numerical difference between these two assessments (3–3 = 0) provides a priority for the site recommendation and LA of "0" for this principal factor.

Integrity of Spent Nuclear Fuel Cladding. *Current Representation in TSPA*–The current cladding integrity model in the TSPA assumes that cladding fails by early failures because of defects at the time of waste acceptance or because of corrosion of the Zircaloy (Volume 3, Section 3.5.1.6). Mechanical failures such as those caused by rockfall have been examined as sensitivity cases. The early-failure fraction is assumed to be 1 percent (range of 0.2 to 11 percent log normally distributed). This accounts for a variety of failure mechanisms that might have occurred in the reactor and during handling (e.g., creep ruptures, hydride reorientation, etc.). It is assumed that these failure mechanisms will not be operative under the cooler temperatures of the repository. The early-failure frequency comes from examination of industry data on in-reactor failures, spent-fuel storage pools, and limited data on dry storage. The Zircaloy corrosion model is based on an assumption that the corrosion rate will be 100 times less than that of Alloy 22 assessed for the corrosion resistant material. Tests by the Naval Nuclear Propulsion Program and others show that under controlled reactor chemistries the general corrosion rates of Zircaloy cladding are extremely low. However, there are uncertainties in the localized corrosion rates of irradiated Zircaloy under the range of conditions that are considered possible in the repository environment. It is assumed in the cladding model that any failure of the cladding along a fuel pin will expose all of the fuel to the external environment.

Significance of Uncertainties to TSPA–The significance of the cladding representation is assessed by considering the presence or absence of cladding in delaying water and oxygen contact with the spent nuclear fuel, rather than looking at the range of uncertainty in the present model (Volume 3, Section 5.5.2). Comparing the peak dose rate without cladding credit, the peak dose rate changes by 1 to 2 orders of magnitude. However, the significance to uncertainty is greater because of the assumptions used to model cladding performance. Accordingly, the assessment of "high" significance is accepted for this prioritization.

Current Confidence–The current confidence in the representation of cladding is judged to be "3." The model may not be realistic or span the range of possibilities. Significant uncertainties exist regarding the early cladding failures and corrosion rates, which make it difficult to assess how realistic the current representation is. Other failure mechanisms, such as mechanical failures, have not been explicitly included in the modeling. There are few data regarding the failure mechanisms and corrosion rates of irradiated Zircaloy under repository environmental conditions. Uncertainties exist regarding the proper early-failure frequency that should be used, whether in-reactor and pool storage data are representative, and regarding the lack of performance data for cladding in spent-fuel dry storage. The potential effects of hydride reorientation or embrittlement over long periods are not considered in the current model.

Confidence Goal–The confidence goal in the representation of cladding is judged to be "5." This reflects the relatively high significance to TSPA, and the desirability of increasing confidence, as well as the capability for doing so. Ongoing corrosion tests are judged to be valuable in constraining the corrosion rate model for Zircaloy. Ongoing studies of cladding failures of fuel rods in dry storage should help to provide better support for the early failure frequencies. It is judged that the defensibility of the cladding model will be important in making a licensing case.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "3," and the confidence goal "5," the numerical difference between these two assessments (5–3 = 2) provides a priority for the site recommendation and LA of "2" for this principal factor.

Dissolution of Spent Nuclear Fuel and Glass Waste Forms. *Current Representation in TSPA*–The spent nuclear fuel dissolution model is an empirically-based model that defines the intrinsic dissolution rate of the waste form as a function of several parameters including temperature, burnup, and carbonate concentration in the water. Because the effects of burnup on dissolution rate appear to be minimal and carbonate concentrations and temperatures are essentially constant after 10,000 years, the dissolution rate is relatively constant at periods beyond about 10,000 years. The dissolution model is based on data gathered under "high flow rate" conditions whereby there is little or no buildup of secondary mineral phases on the spent nuclear fuel or uranium dioxide. Therefore, the model is assumed to provide a dissolution rate, but not necessarily a mobilization rate (Volume 3, Section 3.5.1.7). The mobilization rate is a function of the solubilities of the radionuclides and, for non-solubility limited species, is equivalent to the dissolution rate. It is assumed that solubility-limited species are made available at their solubility limits.

The current model for spent nuclear fuel dissolution does not explicitly account for the formation and radionuclide retardation in secondary mineral phases. Low flow rate and vapor experiments show that secondary phases develop under these conditions and some radionuclides appear to be sorbed onto or absorbed into these phases. There are data to suggest that the solubilities of these secondary phases are six orders of magnitude less than the spent nuclear fuel.

All commercial fuel types are assumed to have the same behavior since data show that there is little dependence on age or burnup of the commercial spent nuclear fuel. DOE fuels have a wider variance in properties and are treated with a single weighted average of those properties to account for that range. The high-level radioactive waste glass dissolution model is based on experimental data on Savannah River glass and is dependent on silica content, pH, and temperature. It is also an empirically based model and incorporates uncertainties in the observational data. The rapid release of radionuclides from the gap between the spent nuclear fuel and the cladding is not addressed because the time steps in the TSPA model are so broad that the pulse release of radionuclides caused by the gap inventory is not discernible in the present predictions.

Significance of Uncertainties to TSPA–Uncertainties in the current representation of dissolution of spent nuclear fuel lead to factors of up to approximately 50X variation in peak dose rate (Volume 3, Section 5.5.1.3). Accordingly, the assessment of "moderate" significance of uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of dissolution of spent nuclear fuel and glass is judged to be "4." The present spent nuclear fuel dissolution model appears to reasonably account for the processes that affect the dissolution rate, but there are uncertainties that appear to lie outside of the range of conditions considered. Secondary phases are not included, so the model is probably not realistic but somewhat conservative. Some uncertainties are not presently represented, such as interaction of spent nuclear fuel and glass chemistries in co-mingled waste packages. Other uncertainties, including the lack of pulse releases in the current model of altered glass caused by vapor phase hydration and calcium concentration, are not considered in the dissolution models.

Confidence Goal–The confidence goal in the representation of dissolution of spent nuclear fuel and glass is judged to be "5." This assessment reflects primarily the likelihood that ongoing tests will significantly increase confidence before submittal of the site recommendation and the LA. Laboratory experiments in progress on spent nuclear fuel under low-flow and water vapor conditions may provide a basis to consider secondary phases, but development of a strong technical basis was judged likely to be incomplete for submittal of the site recommendation and the LA. It is desirable to have a somewhat higher level of defensibility on this principal factor during licensing.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "4," and the confidence goal "5," the numerical difference between these two assessments (5–4 = 1) provides a priority for the site recommendation and LA of "1" for this principal factor.

Neptunium Solubility. *Current Representation in TSPA*–The solubilities used in the current TSPA for several radionuclides were developed through a consideration of data developed under a variety of environmental conditions. However, the solubilities are not modeled as environment dependent, so a broad range of solubilities is used (for example, the range of solubilities used in TSPA for uranium spans six orders of magnitude). The data related to the solubility of neptunium span a wide range and reflect a range of conditions (e.g., temperature) and vary according to whether equilibrium was reached from under-saturated or over-saturated conditions. The neptunium-solubility range used in the current TSPA spans three orders of magnitude, is assumed to be log-uniform, and lies slightly below the data developed from over-saturation and above the data developed from under-saturation (Section 2.2.5.3 of Volume 1 and Section 3.5.1.8 of Volume 3). It is therefore a compromise assessment, which can presumably be improved with additional data. Because it is assumed that temperatures will have returned to ambient levels by the time neptunium solubilities are important, there is no temperature dependence considered in the current model.

Significance of Uncertainties to TSPA–The range of uncertainty in the representation of neptunium solubility leads to factors of up to approximately 50X variation in peak dose rate (Volume 3, Section 5.5.4). Accordingly, the assessment of "moderate" significance of uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of neptunium solubility is judged to be "4." The current range of data on neptunium solubilities includes many data points, which are judged not to be appropriate to the repository conditions. Uncertainties exist as to what the neptunium-controlling phase will be under these conditions. Because of this, the current representation may not be realistic or capture a full range of possibilities.

Confidence Goal–The confidence goal in the representation of neptunium solubility is judged to be "5." Experimental and chemical analyses are currently underway or planned that are readily conducted and will increase confidence. The significance to TSPA is moderate, and it is judged to be desirable to have a more defensible position on neptunium solubility at the time of submittal of the site recommendation and the LA.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "4," and the confidence goal "5," the numerical difference between these two assessments (5–4 = 1) provides a priority for the site recommendation and LA of "1" for this principal factor.

Formation and Transport of Radionuclide-bearing Colloids. *Current Representation in TSPA*–The present representation of colloid formation assumes that there are abundant potential sources of colloids in the engineered barrier system including spent nuclear fuel and glass alteration products, iron oxides from the corroded outer barrier of the waste package, concrete, and natural colloids. The important issues are how much of the plutonium (which is the only radionuclide treated) will be mobilized by colloids and how much will be dissolved (Section 2.2.5.3 of Volume 1 and Section 3.5.1.9 of Volume 3). The ratio of the two is the effective K_c value, and this parameter has been estimated using laboratory solute-K_d experiments for clay, iron, and other colloid types. The current model uses a range of K_c values represented by a log uniform distribution, ranging from 10^{-5} to 10^{1}. This range comes from consideration of column tests and high-drip-rate tests, but there are relatively few tests and few data.

Another significant issue is the reversibility or irreversibility of sorption onto colloids. To simulate the recent findings at the Nevada Test Site where colloids were found to have traveled kilometer-scale distances following a thermonuclear test, colloid-K_ds were assumed to be zero and effective porosities in the saturated zone were assumed to range from 10^{-5} to 10^{-3}. Analyses assuming reversibility of plutonium sorption on the colloids showed no significant effect on peak dose rate. Presumably, analyses assuming no desorption from the colloids would result in somewhat greater dose rates.

Significance of Uncertainties to TSPA–At 100,000 years the peak dose rate has a significant contribution from plutonium mobilized by colloids. By one million years, plutonium is the dominant contributor in 8 percent of the probabilistic realizations (Volume 3, Section 5.5.5). It is judged that if a wide range of effective K_c values and effective porosities in the saturated zone is used, representing bounds on parameter uncertainty, the resulting calculations would show a "moderate" significance to performance especially at long periods. This assessment is accepted for this prioritization.

Current Confidence–The current confidence in the representation of colloid formation and transport is judged to be "2." There is considerable uncertainty as to whether the range of K_c values is realistic and spans the range of possibilities. There are few data, and the applicability of colloid transport observed near an underground thermonuclear test, as opposed to conditions at Yucca Mountain, is very uncertain.

Confidence Goal–The confidence goal in the representation of colloid formation and transport is judged to be "4." It is judged to be reasonable to obtain increases in confidence before submittal of the site recommendation and the LA. Ongoing work on sorption coefficients and on reversibility of sorption onto colloids should be helpful. The Nevada Test Site data are being analyzed, and the Busted Butte tracer tests may be helpful. Additional modeling of transport may also help. Significance to TSPA is moderate and increased defensibility in this principal factor is judged to be needed for the site recommendation and the LA.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "2," and the confidence goal "4," the numerical difference between these two assessments (4–2 = 2) provides a priority for site recommendation and the LA of "2" for this principal factor.

Transport Through and out of the Engineered Barrier System. *Current Representation in TSPA*–In the current representation, advective flow dominates over diffusion within the waste package. All of the exposed waste form is assumed to be in contact with advective pathways. Sensitivity analyses have assumed diffusive pathways over about 20 percent of the waste and show delays in releases of about 7,000 years. Water is assumed to be readily advected from the waste package, there is no "bathtub" effect or associated pulse release, and there is no sorption assumed through the corrosion products of the waste package or concrete from the lining. K_ds are assumed for the invert and the flow is assumed to travel through and mix with a semi-cylindrical section of invert. Pulses occur in association with superpluvial climates and their associated elevation of the water table to quickly mobilize radionuclides into the unsaturated zone (Volume 3, Section 4.1.10).

Significance of Uncertainties to TSPA–This principal factor is directly related to the volume of water that passes through the waste package (Volume 3, Section 5.5.6). By analogy to other factors affecting release of the solubility-limited radionuclides (such as neptunium), which dominate the peak dose rate at later periods, there is "moderate" significance of uncertainty to TSPA. This assessment is accepted for this prioritization.

Current Confidence–The current confidence in the representation of transport through and out of the engineered barrier system is judged to be "3." The representation may not realistically span the range of possibilities. The lack of inclusion of diffusive pathways makes the model unrealistic and perhaps overly conservative. Uncertainties in sorption characteristics and geometries of the invert, and the effect of the chemistry of materials on the drift floor, have not been included. Water is assumed to be readily advected from the waste package, there is no "bathtub" effect or associated pulse release, and there is no sorption assumed through the corrosion products of the waste package or concrete from the lining.

Confidence Goal–The confidence goal in the representation of transport through and out of the engineered barrier system is judged to be "4." It is expected that only modest increases in confidence can be obtained, primarily from some additional data collection and modeling. Experiments and modeling of K_ds for invert and concrete should be helpful, and plans include consideration of diffusive pathways. Some increase in defensibility of this principal factor is desirable before submittal of the site recommendation and the LA.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "3," and the confidence goal "4," the numerical difference between these two assessments (4–3 = 1) provides a priority for site recommendation and the LA of "1" for this principal factor.

Transport in the Unsaturated Zone. *Current Representation in TSPA*–The unsaturated zone transport model uses the flow field from the unsaturated zone flow model directly. A particle tracker has been developed, and accounts for phenomena such as the movement of mass, sorption, and matrix diffusion. Distributions of sorption coefficients (K_d) and matrix diffusion are sampled for each radionuclide and hydrogeologic unit (Volume 3, Section 3.6.1.3). The K_d values come from laboratory studies (Volume 1, Section 2.2.5.3). Two aspects of radionuclide transport are developed from the modeling. The first is the movement of radionuclides from the repository to the water table, resulting in breakthrough curves and spatial distribution for the transported radionuclides. With changes in climate, the distribution of mass at the water table follows the distribution of release with little time delay. The second aspect is the dilution, or the mass of each radionuclide of interest per unit volume of water.

The analysis of transport considers each of the conceptual models currently addressed by the unsaturated zone flow model; that is, the analyses consider 39 calibrated flow fields. The unsaturated zone transport region is divided into six subregions similar in size to the six regions of the unsaturated zone flow model above the repository, having dimensions of about 0.7 by 0.7 km (0.4 by 0.4 mile) (Volume 3, Section 3.6.2). The mass is divided by the flow volume within each region to arrive at the average concentration. This representation is believed to be appropriate assuming that the repository is a large source having dimensions of kilometer scale, which would be the case if all waste packages are breached and releasing radionuclides into the groundwater flow. However, at early times when only a few waste packages have failed, the waste packages would more closely resemble point sources that are small relative to the region size. Thus there is an implied "dilution" just below the repository because of the region size that may not be appropriate for early time when only a few, widely spaced waste packages may have breached.

Significance of Uncertainties to TSPA–Using the high-level TSPA model it is difficult to separate the effects of transport and dilution in the unsaturated zone from these processes in the saturated zone and at the pumping well. Based on limited sensitivity analysis (Volume 3, Section 5.6) and application of judgment, this factor has "high" significance of uncertainty to TSPA. The principal effect is expected to be in the first 10,000 years when there will be only a few waste package failures. A sensitivity test that compares the TSPA base case to a representation of waste packages as point sources, and using the minimal amount of dispersion in the saturated zone assumed for a larger source, and no well dilution, results in differences in peak dose rate of a factor of 1,000. This test represents the extreme results of some conservative assumptions that are similar to the current representation for TSPA. The test is very conservative and unrealistic, but is used to show that the significance of uncertainty to TSPA, for predictive models of unsaturated zone flow and transport, is potentially "high." This assessment is accepted for purposes of prioritization.

Current Confidence–Current information regarding transport in the unsaturated zone is summarized in

Volume 1, Section 2.2.5.3. Concentrations of radionuclides released from the waste packages will be reduced as the contaminated water from the waste packages mixes with uncontaminated water in the matrix and fractures of the unsaturated zone. Because the flow is predominantly in the fractures in welded tuffs, the concentrations may also be reduced as radionuclides diffuse from the fractures into the fine pores of the rock matrix.

Current confidence in the representation of transport through the unsaturated zone is judged to be "2." The current representation may not be realistic in capturing the range of transport behavior during early times when single waste packages may fail. Uncertainties are large and some uncertainties have not been included. The current model may not capture the full range of conditions important to dilution. Also, the current treatment of unsaturated zone transport may be non-conservative at early times because of the assumption of mixing across the entire area of each region as the radionuclides arrive at the water table beneath the repository.

Thermal effects from the repository, such as possible chemical and mineralogical changes to transport pathways for radionuclides, are not included in current models of unsaturated zone flow and transport. Coupled thermal, hydrologic, and chemical processes may increase or decrease the chemical retardation characteristics along transport pathways below the repository. Matrix diffusion along these pathways may decrease because of coupled thermal, hydrologic, and chemical processes. Coupled process effects on fracture-matrix interaction can be predicted using numerical simulators, but there are few applicable test or analog data available.

Confidence Goal–The confidence goal in the representation of transport in the unsaturated zone is judged to be "5." A number of studies are judged to be likely to provide useful information before submittal of the LA, including the Busted Butte experiments and studies in the niches and where the cross drift crosses above the main drift in the Exploratory Studies Facility. These studies are described in Section 3. Additional modeling, with finer gridding and revised properties, will permit more realistic modeling of point sources. Defensibility for models that address unsaturated zone flow and transport is judged to be particularly important for the site recommendation and the LA.

A field tracer test underway at Busted Butte will directly examine geochemical transport in rock units similar to those under the repository. Testing will investigate transport of reactive and non-reactive tracers analogous to key radionuclides in the repository system, and transport of particles that are similar to colloids that can mobilize certain radionuclides. This test will improve understanding of the applicability of laboratory studies to the field. Multiple chemical tracers will be used to discern the effects of solute-specific retardation from dilution caused by mixing with formation water. Because of spatial variability in the characteristics of the tuff units, considerable uncertainty will remain when these test results are used to predict transport conditions in the repository block during the postclosure performance period.

The effects of thermal alteration on the flow and transport properties of the unsaturated zone are not well understood. Progress in this area before submittal of the site recommendation and the LA will consist mainly of development and validation of predictive models for coupled thermal, hydrologic, and chemical processes. Whereas there is substantial uncertainty with respect to the chemical retardation characteristics of the unsaturated zone, dilution behavior may be more readily interpretable for the altered system.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "2," and the confidence goal "5," the numerical difference between these two assessments (5–2 = 3) provides a priority for site recommendation and the LA of "3" for this principal factor.

Flow and Transport in the Saturated Zone. *Current Representation in TSPA*–The saturated zone flow and transport model begins with the distribution of flow and mass from the six regions in the unsaturated zone model (Volume 3, Section 3.7.2). These regions are continuations of the regions considered in the unsaturated zone transport analyses. These six regions are then modeled as "flow tubes" through the saturated zone, which define a plume at a down-gradient receptor well, where it is assumed that the well extracts from the area of highest concentration within the plume. Groundwater flux beneath the site is assessed to be about 0.5 m/year (2 ft/year), based on the hydraulic properties of the volcanic aquifer and a typical hydraulic gradient estimated from regional gradients. The regional groundwater flow model with a grid spacing of 1,500 m (4,900 ft) is used to determine flow directions and flow paths within three hydrologic layers (Volume 3, Section 3.7.1). The last portion of the flow path is considered to be in an alluvial aquifer, which contributes between 0 and 6 km (0 and 4 miles) to the 20-km (12-mile) flow path.

Some factors (e.g., saturated zone dilution) were developed from expert elicitation and process model abstraction. To some degree, expressions of current confidence may reflect judgment on the TSPA methodology rather than the state of knowledge of expected transport behavior. For example, within each flow tube in the saturated zone used for TSPA, dispersion processes operate in the model but are very limited, especially the vertical transverse dispersivity. The dilution factor at the maximum concentration part of the plume at a distance of 20 km (12 miles) is sampled from a lognormal distribution ranging from 1 to 100, with a mean of 3. This conceptualization is based on opinions from the saturated zone expert elicitation panelists, and is not necessarily consistent with available data or other conceptual models.

Significance of Uncertainties to TSPA–Analysis of the effect of uncertainties in the saturated zone transport representation, holding the unsaturated zone transport and pumping well dilution constant, show a "moderate" significance of uncertainty to the TSPA (Volume 3, Sections 4.3 and 5.7). In the first 10,000 years the influence will be "moderate" because of the dry climate and more limited water fluxes and consequent radionuclide releases. In later times it will also be "moderate" because of the greater water fluxes and increased dilution. This assessment is accepted for the prioritization.

Current Confidence–The current confidence in the representation of flow and transport in the saturated zone is judged to be "2." This relatively low rating reflects the likelihood that the current representation does not realistically span the range of possible interpretations of the available data. The current representation in TSPA is conservative in its consideration of dilution, but there are few data on which to base other estimates. A characterization data gap exists at flow path distances of 5 to 20 km (3 to 12 miles), which is a relatively large portion of the potential radionuclide transport pathway.

Transient conditions (on a time scale of thousands of years) are not accounted for in, and have not been incorporated into, the model and hydrochemical data. The present model is steady state and does not account for transient changes in the hydraulic gradient caused by climate change or pumping wells, which could lead to increased dilution. Hydrochemical data suggest that the groundwater flux beneath the site may be smaller than assumed for present conditions, but this possibility has not yet been included. In addition, sorption behavior and chemical precipitation along transport pathways with reducing conditions could be quite different from those included in the model, and there is some evidence that such conditions exist. Finally, the conceptualization of flow in stream tubes may not be appropriate and there is a lack of treatment of matrix diffusion.

Confidence Goal–The confidence goal in the representation of flow and transport in the saturated zone is judged to be "3." Several studies are ongoing or planned that can potentially provide additional confidence to this principal factor. Cooperative studies planned with Nye County will provide information on hydrostratigraphy in the various aquifers, and improvements to the hydrogeologic framework model will provide information on flow paths. Boreholes SD-6 and WT-24 may provide additional information on flow beneath the site and analysis of existing hydrochemical data may help in assessing flow conditions in the present and wetter climates. Examination of transport processes for other, large-scale, analogous contaminant plumes may be useful. A refined model of saturated zone flow and transport will be developed for the site recommendation and the LA that incorporates this additional information, which will increase confidence in the TSPA. However, the confidence goal assessment is tempered by constraints on what can be done to further characterize this relatively large area.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "2," and the confidence goal "3," the numerical difference between these two assessments (3–2 = 1) provides a priority for site recommendation and the LA of "1" for this principal factor.

Dilution from Pumping. *Current Representation in TSPA*–The present model of dilution is simple and conservative. It assumes that the receptor well extracts from the highest concentration portion of the plume and that the low flow rates for a typical domestic well produce no additional dilution (Volume 3, Section 5.3). The well is assumed to lie within a relatively large contaminant plume and not within a narrow plume that might allow for dilution from the mixing and extraction of uncontaminated water. The current dilution model assumes that essentially all dilution will be obtained in the unsaturated zone before the radionuclides reach the saturated zone.

Significance of Uncertainties to TSPA–One measure of the significance to TSPA would be to assume no dispersion in the unsaturated zone or the saturated zone, such that the contaminant plume would be very narrow, unlike the current representation in which dispersion is significant for the unsaturated zone. This results in a factor of up to approximately 50X change in the peak dose rate (Volume 3, Section 5.8.2). The assessment of "moderate" significance of uncertainty is accepted for this prioritization.

Current Confidence–The current confidence in the representation of dilution from pumping is judged to be "5." Because of constraints imposed by the requirement to consider the maximum concentration within the plume, there is no uncertainty as to whether the well penetrates this part of the plume, or in the dilution that might occur because of typical mixing of uncontaminated waters in the well. Confidence is not higher because this representation may be overly conservative and unrealistic.

Confidence Goal–The confidence goal in the representation of transport in the unsaturated zone is judged to be "5." While believed to be a conservative representation, little can be done to increase confidence in the representation that would likely be acceptable for reducing this level of conservatism.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "5," and the confidence goal "5," the numerical difference between these two assessments (5–5 = 0) provides a priority for site recommendation and the LA of "0" for this principal factor.

Biosphere Transport and Uptake. *Current Representation in TSPA*–The biosphere model uses the radionuclide concentrations that are extracted from the wells, and dose conversion factors, as input to arrive at dose from unit concentrations. To establish site-specific dose information, a variety of domestic users have been considered, including a residential farmer, subsistence farmer, and an "average" resident. The overall biosphere model, which has been used for many studies, includes uncertainties in several key components such as drinking water versus water for irrigation, plant uptake, resuspension, and livestock uptake (Volume 3, Section 3.8.1).

Significance of Uncertainties to TSPA–Sensitivity tests show that the uncertainties in the biosphere model lead to variations in peak dose rate factors of approximately 5X (Volume 3, Sections 4.3.2 and 5.8.1). Because this factor represents a model that is widely used for dose assessment, an assessment of "low" significance of uncertainty to TSPA is accepted for this prioritization.

Current Confidence–The current confidence in the representation of biosphere transport and uptake is judged to be "5." The model realistically bounds the range of possibilities and uses generally accepted conceptual models. Some of the assumptions in the model will need to be defended at the time of licensing. Concerns have been raised about doses to children and the uncertainties in site-specific soil properties, which presently are being addressed.

Confidence Goal–The confidence goal in the representation of transport in the unsaturated zone is judged to be "5." Ongoing studies are addressing some of the uncertainties in the present model. Dose to children under the age of 6 years should be considered to increase confidence in model exposures of a sensitive subpopulation. In addition, sorting the data sets for inclusion of desert environments could increase confidence in the current representation. The relatively low significance to performance tempers the desire to focus further on increasing confidence in this principal factor.

Priority for Site Recommendation and LA–Based on the assessment of the current confidence "5," and the confidence goal "5," the numerical difference between these two assessments (5–5 = 0) provides a priority for site recommendation and the LA of "0" for this principal factor.

2.2.5 Performance of Design Options

Design options and alternatives that could improve performance and reduce the sensitivity of repository performance estimates to uncertainties in the repository system will be evaluated in a comprehensive assessment to support initial site recommendation and LA design selection in 1999. Preliminary evaluations have already been conducted for the design options that were developed for the VA reference design, which are described in Volume 2, Section 5.3.

Sensitivity analyses for performance of backfill in the emplacement drifts, drip shields, and ceramic coatings to the waste package are reported in Volume 3, Section 4.5. Those studies indicate the potential for improved postclosure performance and reduced sensitivity of performance estimates to uncertainties associated with TSPA process models. With reliance on these design options, the analyses also show that defense in depth would be enhanced as an element of the postclosure safety case. However, the analyses are preliminary and some issues related to implementation and longevity of the design options have yet to be resolved. Potential benefits and liabilities of these options are discussed in Volume 2, Section 5.3. The following sections summarize the potential importance of the VA design options, and the associated issues for which technical work is needed and will be undertaken if warranted from the initial site recommendation and LA design selection decision in 1999.

2.2.5.1 Backfill

Backfill could provide mechanical protection from the effects of eventual drift collapse for the waste package and other engineered barriers. 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 prevent dripping water from directly contacting the waste packages. Performance assessments described in Volume 3, Section 4.5.1, indicate that, under the assumptions used for TSPA-VA, the thermal effects of backfill would have little benefit to postclosure performance. More complete analyses of backfill effects on seepage are planned for the site recommendation and LA reference design selection in 1999. These analyses may show additional benefits to performance.

2.2.5.2 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 beneath and above a drip shield would protect it from mechanical damage. In addition, backfill could inhibit diffusive transport of radionuclides beneath the drip shield. Performance assessment sensitivity studies (Volume 3, Section 4.5.2) indicate that the backfill-drip shield combination would be highly effective for reducing the number of breached waste packages.

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., corrosion-resistant metal alloys) and stability of the backfill properties. Second, flow diversion performance of the backfill and drip shield combination needs to be demonstrated for a wide range of flow conditions such as could occur in the repository over many thousands of years. Finally, although there do not appear to be conceptual problems in implementing this barrier, engineering feasibility analyses have not yet been performed.

2.2.5.3 Ceramic-Coated Waste Package

Ceramic coatings on metals have been observed to be long-lasting and impervious to water. Ceramic materials can be applied using available industrial practices and may substantially prolong the waste package lifetime. Application of ceramic technology to engineered barriers, including the waste package, is discussed in Section 5.3.3.

It is likely, based on general chemical and physical properties of ceramics, and observations of natural and man-made analogs, that such coatings on the waste packages would remain intact for one million years or longer (backfill may be required to mitigate the effects of rockfall). Such a barrier would have a significant effect on performance as long as it remains intact, and would increase the lifetime of the corrosion-resistant material used as the substrate for the coating. This approach would also be highly effective at reducing the number of waste package failures and would provide defense in depth. Preliminary analysis of the performance contribution from ceramic coating on the waste packages is described in Volume 3, Section 4.5.3.

There are several significant uncertainties with this option. Stability against ceramic 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 contribute to corrosion. 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. Technical work in these areas will significantly increase confidence in the representation of ceramic coatings in performance assessment.

2.2.5.4 Allocation of Performance

Performance has not yet been allocated to the different features of the repository system design, because that design has not yet been selected. The prioritization of technical work related to postclosure performance is conditional, because it is based on the VA reference design. The TSPA inputs considered in the prioritization process, for example, were generated using models based on the VA reference design.

The repository system design selected for the site recommendation and LA decisions may differ from the VA design. Accordingly, when the initial design selection is made in 1999, the principal factors affecting postclosure performance will be redefined and another prioritization will be conducted. The reprioritization will take into account not only the design features, but additional information obtained by that time and the decision-making objectives established for selection of the initial site recommendation and LA design. That comprehensive design selection process is discussed in more detail in Section 2.4.3. Postclosure performance considerations, including the degree of design margin and defense in depth, will be key considerations in that selection process.

2.2.6 Summary of Technical Work to Complete the Postclosure Safety Case

This section briefly summarizes the prioritized technical work to address the principal factors of postclosure performance and the planned work to address the other elements of the postclosure safety case. More detailed work plans in each of these areas are provided in Section 3.

2.2.6.1 Summary of Technical Work to Address Element 1: Assessment of Expected Postclosure Performance and Supporting Evidence

Performance assessment models and abstractions will be updated in the next year to reflect updated process models, and to incorporate the results of the design alternatives evaluated in 1999. Models and abstractions will be selected for the TSPA for the LA in August of 1999. Those models will be used in the analyses to support the site recommendation and LA decisions.

The updated TSPA models and methods will also be applied to support the interpretation of results from the cross drift characterization activities, to support the TSPA for the LA in 1999, to support site suitability evaluations for the site recommendation in November 2000, and to support the acceptance draft of the LA in February 2001.

Technical work to update the supporting models will be conducted primarily to address the principal factors that are assigned relatively high priority in Section 2.2.4 (Table 2-2). These include:

- Drift seepage and percolation to depth
- Effects of heat and excavation on flow
- Dripping onto waste packages
- Chemistry of water on the waste packages
- Integrity of the inner corrosion-resistant waste package barrier
- Integrity of the spent nuclear fuel cladding
- Formation and transport of radionuclide-bearing colloids
- Transport in the unsaturated zone

The planned work in these areas is summarized in the following paragraphs. More detailed information about this work, and other planned work to address principal factors with lower priority, is provided in Sections 3.1 and 3.2.

Drift Seepage and Percolation. Data on percolation and seepage at the drift scale will continue to provide insights on the processes that will control the amount of water that may contact waste packages. Planned work will examine percolation over that part of the repository layout accessed by the cross drift and in geologic layers that will include more of the repository horizon. Plans to support the site recommendation and the LA are detailed in Section 3.1.2.3 and include the following testing and modeling activities:

- **Excavate two additional niches and prepare two fracture/matrix test beds in the cross drift.** Seepage and fracture/matrix interaction tests will be performed in the lower (lithophysal and non-lithophysal) Topopah Spring units that comprise the majority of the potential repository horizon. Planned tests include liquid release tests and long-term tracer injection tests.

- **Perform additional geochemical and isotopic analyses to determine where water has flowed in the past.** Measure the concentrations of chemical components in the rock such as chloride, bromide, and sulfate, and use the results to identify fast paths and travel times. Ongoing analyses of the isotopic ages of fracture-lining minerals will provide information on the history of water movement through the site. These studies show how and when water has moved through the unsaturated zone and reveal characteristics of the water, such as the chemical composition and temperature.

- **Perform a controlled study of percolation from the cross drift to the underlying Exploratory Studies Facility main drift.** The cross drift infiltration experiment in the crossover alcove will provide data on percolation rates through fractured welded tuffs under controlled boundary conditions.

- **Monitor moisture conditions in the Exploratory Studies Facility including the cross drift.** Moisture monitoring activities in Alcoves 1 and 7 of the Exploratory Studies Facility will continue and monitoring in the cross drift will be established, for study of moisture balance, ventilation effects, and the movement of water used in construction.

- **Update percolation and seepage models.** Percolation processes have been modeled at two scales: mountain and drift. These models will be updated so that the site hydrostratigraphy is consistent with recent laboratory and field test data. Models for seepage into drifts will also be updated to encompass field test data and the effects of thermally driven coupled processes on seepage, as discussed in the following paragraphs.

The effects of heating on seepage are being investigated in a drift-scale thermal test that is presently underway, and by laboratory experiments that will support models for predicting the effects of coupled processes over much longer periods.

Effects of Heat and Excavation on the Flow System. Planned testing, experiments, and analysis will continue to address thermally driven changes that are expected to occur in the repository host rock. This principal factor will be addressed by several types of planned technical work including the following:

- **In situ thermal testing.** Thermal testing will be conducted until and beyond the site recommendation and the LA. The large block test and the single heater test will be completed with post-test analysis and documentation. The drift scale test will continue heating for 4 years, then cool for a similar period, with data collection and analysis throughout. A thermal test will also be conducted in the cross drift, but only preliminary results will be available in time to support the site recommendation and LA decisions. Field thermal tests will continue to provide much needed physical and chemical data on the response of the host rock to heating, for validation of predictive models of repository performance. More detailed description of planned thermal tests is provided in Section 3.1.4.

- **Mineral alteration studies.** Work will be conducted to determine, through modeling and experiments, the extent to which drainage of warm water will alter tuff layers below the repository. In particular, the planned work will evaluate whether the lowermost unit of the vitric tuff layers below the repository may become less permeable and divert water above the Calico Hills tuff unit. Planned work will also address the formation of minerals in fractures above the emplacement drifts that could decrease the permeability and inhibit seepage. In addition, experiments will be conducted to evaluate how fracture-matrix interaction along potential transport pathways will be permanently altered. This work is described in more detail as part of the transport model development in Section 3.1.2.5 and coupled process studies in Section 3.1.5.

Predictive models of rock response to repository heating will be updated to reflect new information. Modeling work will include recalibration based on moisture monitoring and in situ thermal test results, revision of seepage calculations, calculating the effects of a silica/calcite cap that may form over the drift, and modification of seepage models to include the geometric effects of rockfalls and collapse.

Dripping onto Waste Packages. Various design options such as backfill and drip shields have the potential to mitigate the quantity of water that may drip onto waste packages. The planned work will emphasize engineered measures to control dripping onto waste packages, and will be based on both predictive modeling and experiments (Section 3.2.1.12). Engineered barrier testing will be performed in laboratory settings, and in larger scale proof of principle testing arrangements. Predictive models will be confirmed by the testing, and preliminary results will be obtained in time to support site recommendation and LA decisions. It is expected that design options to control dripping onto waste packages will significantly enhance expected performance, defense in depth, and other aspects of the postclosure safety case.

Chemistry of Water on the Waste Packages. The chemistry of water contacting waste packages may influence the rate of corrosion that leads to breaching of the waste packages, and the rates of radiocnuclide release from the waste forms inside the waste packages. Planned technical work will provide additional information on bulk water chemistry in the near-field, which is used as a chemical boundary condition for corrosion and radionuclide release models. The following additional work is planned to support the site recommendation and the LA:

- **Monitor and evaluate water, mineral, and gas samples from field thermal tests.** Field tests have already provided observational data on coupled processes, including the evolution of water chemistry. These tests constitute direct physical and chemical models of the repository environment for the time scales covered by the tests. It is expected that results from the drift scale test will be important in substantiating predictive model capabilities. The planned work for field thermal testing is described in Section 3.1.4.

- **Validate predictive models of coupled processes that control water chemistry.** Perform laboratory experiments and modeling simulations to validate predictive models of coupled processes that control water composition. Recent studies have shown that coupled processes may progress slowly on time scales that are not readily accessible by field testing. Predictive simulation will therefore be relied on to understand some of the long-term impacts to barrier performance. Controlled laboratory investigation of coupled processes will be emphasized to support model validation for this purpose. In addition, gas-phase processes will be simulated using models which have been tested against observations from the field thermal tests. These tests are included in the thermal, hydrological, and chemical coupled process studies described in Section 3.1.5.5.

- **Conduct testing and modeling of introduced materials.** Materials introduced to the near-field environment, such as steel and concrete, will affect water chemistry. Planned technical work will include, but not be limited to, physical and chemical stability to heating, permeability, leachate chemistry, colloid generation, radionuclide retardation, and the effects of limiting nutrients conducive to microbial growth. The planned work for comparative testing of introduced materials is described in Section 3.1.5.2.

Planned technical work will update mechanistic chemical models for processes that control the composition of the water contacting the waste packages and other engineered barriers and the effects of introduced materials in the near-field geochemical environment. The behavior of reactants in the near-field geochemical environment will be coupled with thermal-hydrologic behavior and mountain-scale convective processes.

Waste package corrosion may be strongly influenced by micro-environments on the package surface which are associated with films, pits, or crevices that may form as a result of abiotic or microbially mediated processes. These mechanisms will be investigated in connection with long-term material testing for waste package design.

Integrity of Inner Corrosion-Resistant Waste Package Barrier. Planned technical work to address this principal factor will emphasize corrosion rates under conditions that might occur in crevices and the nature of phase transitions near grain boundaries. These objectives will be addressed by the following technical work:

- **Corrosion testing.** Long-term corrosion tests for representative repository conditions will continue to be conducted for high-nickel and titanium-based alloys. Specimens will be exposed in the vapor phase (humid conditions), at the water line (partially immersed), and fully immersed (underwater). The interaction between the corrosion-resistant and other materials when the waste package is cracked or breached will also be assessed. These activities are described in more detail in Section 3.2.2.9.

- **Model for localized corrosion.** Measurements will address conditions that lead to localized corrosion (pitting, crevice corrosion, stress corrosion cracking, and hydrogen embrittlement initiation) on susceptible materials. The critical potentials for passive film breakdown and re-formation (or repassivation) will be measured over a wide range of temperature, pH, and electrolyte compositions. These tests and measurements are described further in Section 3.2.2.9.

- **Model for phase stability and effects.** Testing will be conducted to determine when thermal aging causes phases that can degrade performance. Thermal aging could impact a material's corrosion behavior, as well as its mechanical response to rockfall and other applied loads. This work will characterize the microstructure of materials and any changes that may occur. Samples of as-fabricated and as-welded container materials from fabrication mockups will be used. This planned work is described further in Sections 3.2.2.6 and 3.2.2.9.

- **Microbially induced corrosion.** Experiments will continue to identify conditions for which microbial activity could hasten corrosion of the waste package inner barrier. Limiting factors for growth of microorganisms in the near-field environment, and the nature of microbial interactions with corrosion-resistant materials, will be further evaluated by experiments and the results will be used to refine predictive models. These testing and modeling activities are discussed further in Section 3.2.2.9.

More details of the specific work plans to support site recommendation and the LA are provided in Section 3.2.2. In particular, this section describes the design activities for waste packages to accommodate different types of waste. Long-term testing of materials for the waste package and other engineered barriers for performance confirmation will be conducted until and beyond submittal of the LA.

Integrity of Spent Nuclear Fuel Cladding. A preliminary review of models for cladding degradation has identified the need for data on specific parameters of the model. The testing program will provide most of the needed parameters by means of short-term and long-term testing of cladding sections. In addition, the model for the cladding failure based on the behavior of Zircaloy will be refined. Several mechanistically different cladding failure modes will be evaluated for a minimum lifetime-to-failure model. This, in turn, will support more definitive allocation of postclosure performance requirements to the fuel cladding.

Planned work on cladding performance will be conducted as part of an integrated investigation of waste form materials that will continue to support development of appropriate models to simulate waste form alteration and oxidation, and radionuclide release. These tests will include long-term oxidation, dissolution, surface alteration, and radionuclide release rates for spent nuclear fuel (including high-burnup commercial fuel and representative DOE spent nuclear fuels) under a range of wetting modes. Spent nuclear fuel and high-level radioactive waste glass will be tested under conditions anticipated for the repository, to provide data on dissolution and release rates for film flow and dripping water conditions. For example, in one test spent nuclear fuel, both in cladding and crushed in a thin film, will be tested for cladding integrity and effects, alteration, and radionuclide release rates. Performance data for spent nuclear fuel in dry cask storage will be obtained by planned studies of early fuel rod failures in dry storage. Planned testing of waste form and cladding characteristics and behavior is discussed further in Section 3.2.2.8.

Formation and Transport of Radionuclide-Bearing Colloids. Planned work to address this principal factor will include investigation of colloid formation and mobility, affinity of colloidal particles for radionuclides, formation of colloids from waste form alteration, and solubility constraints on colloid formation. Several types of technical work will be performed, emphasizing experiments and observation of colloidal processes in nature:

- **Laboratory studies of the formation and stability of radionuclide-bearing colloids.** Additional laboratory work will be performed to address uncertainties about the formation and stability of colloids, which are likely to form at repository conditions, and the nature of their interaction with radionuclides. The relationship of solubility limits to colloid formation will be investigated further and the reversibility of radionuclide sorption to colloidal particles will be addressed. Planned work on colloid formation is discussed in Section 3.1.2.1.

- **Testing of candidate introduced materials.** The tendency for colloids to form during the alteration of introduced materials, such as steel and concrete, and the affinity of those colloids for radionuclides will continue to be evaluated experimentally. Test results will be incorporated into performance assessment models and will be used for comparative evaluation of materials selected for use in the repository. Testing of introduced materials is described in more detail in Section 3.1.5.2.

- **Observation of colloid mobility.** Movement of artificial colloids through fractured, partially saturated tuff will be investigated in the ongoing field test at Busted Butte, which is discussed further in Section 3.1.2.2.

- **Reviews of formation and stability of radionuclide-bearing colloids at natural analog sites.** Laboratory measurements will be compared with observations from other sites, including the Nevada Test Site, Peña Blanca in Mexico, and research facilities in the United States, such as Oak Ridge National Laboratory and the Idaho National Engineering and Environmental Laboratory. Analog studies of radionuclide transport, which will include observations of colloid behavior, are included in the planned work described in Section 3.1.2.2.

This principal factor is addressed by planned technical work in several different areas of the project including geochemical investigations for unsaturated flow and transport, engineered barrier testing, and waste form testing.

Transport in the Unsaturated Zone. Technical work planned to address this principal factor will emphasize additional experimentation to update the transport model. Colloid formation and transport will be investigated as discussed previously. Advective and diffusive transport characteristics, including those caused by fracture matrix water interaction, will be investigated primarily by field testing. The planned work can be summarized as follows:

- **Experimental evaluation of unsaturated zone transport processes.** Field and laboratory tests will be used to further reduce uncertainties associated with transport of radionuclides in the unsaturated zone. Field testing will continue at the Busted Butte test facility, to observe the transport of radionuclides and colloids in the Calico Hills tuff unit. Planned experiments in the Exploratory Studies Facility cross drift will examine the transport of multiple tracers in the host rock. Planned laboratory tests will evaluate changes in fracture-matrix interaction that may result from coupled processes in the thermally altered zone. Results from field and laboratory testing will be used to update the process models, which are the basis for treatment of unsaturated zone flow and transport in the TSPA. This planned work is part of the testing and modeling discussed further in Sections 3.1.2.2, 3.1.2.3, and 3.1.5.5.

- **Natural and man-made analogs.** Planned work will review observations of radionuclide transport, as dissolved and colloidal species, at analog sites including former weapons facilities where environmental restoration is underway. The review will be conducted with a view to testing the process models developed for Yucca Mountain. This planned effort is part of the technical work described in Section 3.1.2.2.

- **Mountain-scale model of unsaturated zone flow and transport.** The current numerical model will be updated to reflect new information obtained from field testing. This work will include updating parameter values to incorporate understanding gained from field testing at Busted Butte, from in situ thermal testing, from hydrologic tests in the cross drift, and from laboratory measurements. In addition, the model will incorporate improved representation of the geologic units through which unsaturated zone transport would occur and more realistic representation of fault zones and spatial variation in hydrogeologic properties. This work is discussed further in Section 3.1.2.4.

Understanding of unsaturated zone flow and transport is expected to increase significantly before submittal of the site recommendation and the LA, primarily because of the emphasis on field testing. Multi-year field tests, such as the ongoing test at Busted Butte and the imminent tests in the cross drift, have been in planning and preparation for years and are expected to substantially reduce uncertainty associated with transport.

Other technical work to be performed at a lower level of support includes work for the principal factors including precipitation and infiltration, dissolution of spent nuclear fuel and glass waste forms, neptunium solubility, transport through and out of the engineered barrier system, and flow and transport in the saturated zone. Planned work to address these factors is described in Section 3.

Flow and Transport in the Saturated Zone. Saturated zone data collection and process modeling will support refinement of the model used in TSPA for saturated zone flow and transport. Plans to support the site recommendation and the LA include the following:

- **C-wells hydraulic testing**. Hydraulic testing at the existing C-wells complex of boreholes will be completed. Hydraulic and tracer testing will provide hydraulic parameters and sorption information from depth intervals that have not yet been extensively tested.

- **Saturated zone hydrochemistry**. Hydrochemistry data will be obtained using samples collected from existing saturated boreholes at or near Yucca Mountain. In cooperation with Nye County, the USGS, and DOE/Nevada Operations, additional samples will be taken from other wells, including wells in Amargosa Valley and on the Nevada Test Site.

- **Early warning drilling program (Nye County)**. Nye County, Nevada, in cooperation with DOE, will implement a program to install and monitor a series of wells in the Amargosa Valley and the southern Nevada Test Site. The purpose of the program is to monitor the saturated zone along possible transport pathways from Yucca Mountain for radionuclides that would be present in the inventory of the potential repository. The program will consist of shallow and deep wells. The shallow wells will provide hydraulic parameters for the alluvial and tuff aquifer and permit detection in the shallow waters. The deep wells will provide additional data on aquifer properties for the carbonate rocks which underlie Yucca Mountain and, in the future, will serve to detect any contamination that might migrate from the potential repository into the volcanic or carbonate aquifers.

- **Site-scale saturated zone flow and transport model**. The site-scale model, which simulates the movement of water and radionuclides, will be refined to include recent data including regional hydrostratigraphic data south of Yucca Mountain, hydraulic and transport testing a the C-Wells and other existing boreholes at Yucca Mountain, and regional hydrochemistry and isotopic data. The model will be refined to address existing assumptions and simplifications that result in highly conservative predictions of transport. Performance assessment for the site recommendation and the LA will be supported by model abstraction activities, sensitivity studies, and documentation of modeling results. The results of the flow and transport modeling will support Milestones M2MD and M2MF (Section 7).

- **Regional-scale saturated zone flow model.** The regional flow model will be updated to include new and revised regional hydrostratigraphic data including data from south of Yucca Mountain, hydrostratigraphic data from the Nye County boreholes and existing boreholes at Yucca Mountain and the

Nevada Test Site, and stratigraphic information from existing geophysical data.

- **Downgradient hydraulic and tracer testing.** In cooperation with Nye County, new wells will be drilled to evaluate hydraulic parameters, effective porosity, longitudinal dispersivity, colloid transport parameters, and sorption characteristics of alluvium and volcanic rocks downgradient from Yucca Mountain. Hydraulic testing and tracer tests will be designed to investigate potentially important strata and flow paths based on new information as it is collected.

Performance Assessment Work. Performance assessment work in preparation for submittal of the site recommendation and the LA will include incorporating new site data, updating the process models and abstractions, and evaluating information on natural and man-made analogs. Process models will be updated to include advances in scientific understanding in time to support selection and abstraction of the models in August of 1999. Performance assessment abstractions and models will be updated in response to new information and independent external review. Updated TSPA capabilities will support the site recommendation and LA decisions.

Updated performance assessment models and methods will also be applied to specific evaluations. These applications include support to the design alternatives evaluation in 1999, interpretation of results from the field tests, and support to the site suitability evaluations for the site recommendation, as well as the TSPA for licensing. Activities associated with this work are identified and described further in Section 3.3.

2.2.6.2 Summary of Technical Work to Address Element 2: Design Margin and Defense in Depth

This section describes postclosure performance analyses and associated testing activities that will be conducted specifically to support eventual selection of the design. The full scope of design work leading to the selection of the design for the site recommendation and licensing decisions is described in Section 3.2. The approach to alternatives evaluation and an initial set of alternative design concepts are presented in Volume 2, Section 8.

Technical work will be conducted to support analysis of postclosure performance for the evaluation of design alternatives and design options. This work will be completed in time to support selection of the initial site recommendation and LA design in May of 1999 and the final design in November 2000. It will include evaluation of the postclosure behavior of materials, analysis of the technical feasibility of design alternatives, and proof of principle testing. This planned work is described in Sections 3.1.5.1, 3.2.2.9, 3.2.1.12, and 3.3.

Comparative testing of candidate materials will be conducted where needed to support material selection (Section 3.1.5.2). Testing will include evaluation of time-dependent changes (e.g., concrete alteration). Material sources will be reviewed and source specifications compiled for repository construction and engineered barrier materials. Natural and man-made analogs to material behavior will be identified and evaluated for use in performance analysis.

Technical feasibility analyses will be conducted to evaluate postclosure performance of alternative design features. Design analyses, described in Sections 3.2.1 and 3.2.3, will consider fabrication, emplacement, and preclosure performance issues to the extent that these affect postclosure system performance. Predictive modeling will be used to evaluate the postclosure performance impacts of alternative repository layouts, thermal management concepts, ventilation options, timing of repository closure, and the performance of engineered barriers (Sections 3.1.5.1 and 3.2.1.12). Additional performance measures will be developed for TSPA, as needed to capture the important differences between alternative barrier concepts.

Key features of the design will be evaluated using bench scale or larger-scale proof of principle testing. For some design features it is expected that technical feasibility may be established through the use of prototype evaluations. Testing activities to support this effort are discussed in Section 3.2.1.12.

The planned evaluation of design alternatives will include the VA design options as a starting point (Volume 2, Section 5.3). Information needs for evaluating these options include flow properties of backfill, longevity of materials under repository conditions, and demonstration of technical and engineering feasibility. Several types of technical work are planned specifically to address the VA design options:

- **Models for drip shields and backfill.** These models represent the performance of engineered barrier system design options including drip shields and backfill. The work will evaluate sensitivity to uncertainty in hydrologic properties of engineered barrier system components and the host rock, hydrologic boundary conditions, and degradation of engineered barrier system materials over time.

- **Backfill behavior.** Thermal, hydrologic, and transport properties of alternative backfill materials will be measured at unsaturated conditions. Candidate backfill materials will also be subjected to chemical alteration under hydrothermal conditions similar to what might occur in the near-field environment.

- **Performance of ceramic materials.** Ongoing studies of candidate drip shield materials will continue to support and confirm the material selection decisions made in the design alternatives evaluation. Industrial experience will be reviewed, and further tests of ceramic materials will be performed as necessary. Natural or man-made analogs will be used to the extent practicable for assessing the longevity of ceramics and other man-made materials. Ceramic materials will be evaluated as stand alone materials (e.g. ceramic drip shields) and as coatings (e.g., ceramic coatings to waste package barriers). Further tests will be conducted to evaluate the stability of ceramics against phase transitions, the reliability of spray coating methods, the permeability and density of coatings, the long-term effects of metal substrates, and the long-term continuity of coatings. Planned testing will also determine the adhesive strength of coatings and their ability to withstand thermal and handling loads. Corrosion tests will be performed to evaluate the effects of ceramic thickness, structure, and composition.

More detailed description of this planned work is provided in Section 3.2.1.12. Results from these activities will be combined with other information on alternative design features in the comprehensive evaluation of design alternatives to be conducted in 1999. Design margin and defense in depth analyses will be a central part of that evaluation.

2.2.6.3 Summary of Technical Work to Address Element 3: Consideration of Disruptive Processes and Events

The planned technical work for disruptive processes and events consists mainly of completing the development of methods to evaluate the possibility of postclosure nuclear criticality and consolidating and documenting current information. Current information on seismic processes, the effects of volcanism, and inadvertent human intrusion for natural resource extraction appears to be adequate to address postclosure performance issues (Volume 1, Section 2.2.7).

Technical work regarding postclosure nuclear criticality is nearly complete, although additional information is needed to address the following aspects of radionuclide transport and criticality analysis: formation and stability of colloids bearing fissile radionuclides, transport of colloids in the near-field and altered zone, and methodology for addressing criticality in the regulatory arena. These needs will be addressed by planned laboratory work to evaluate colloid formation from spent nuclear fuel interaction with water and to investigate colloid interaction with, and mobility through, introduced materials in the near-field environment. In addition, an improved general understanding of colloid mobility in the unsaturated zone will result from the field transport test at Busted Butte. All of these radionuclide transport issues are addressed in the previously noted technical work plans. Additional work planned to address the remaining criticality methodology questions is discussed in Section 2.4.1.

2.2.6.4 Summary of Technical Work to Address Element 4: Insights from Natural and Man-Made Analogs

Natural analog studies can provide information to help interpret the geologic and hydrologic conditions at the Yucca Mountain site, and the evolution of site conditions associated with future climate changes, repository heating, and introduced materials. Similarly, man-made analogs may provide information regarding relevant processes or materials over time scales and distances that cannot be readily reproduced in a testing program. The behavior of relevant materials and systems has been studied directly in a wide variety of natural and man-made settings. Much of the planned work in this area will consist of review and application of existing information.

Information is needed for natural analogs to the following aspects of repository performance assessment; solubility or colloid facilitated transport of radionuclides, stability and radionuclide retardation characteristics of alteration minerals in fractures, infiltration conditions in the shallow unsaturated zone, and behavior of man-made materials. Studies of colloid transport in the field are particularly important because information from various sites where radionuclide migration in groundwater has been observed directly suggests that the issue could be significant. Planned work will include review of results from such sites for relevance to Yucca Mountain.

To prepare for the site recommendation and LA decisions, a comprehensive review and summary of analog information relevant to performance of a Yucca Mountain repository will be compiled. This review will include analogs for radionuclide solubility and geochemical processes that affect transport, such as the Peña Blanca site in northern Mexico, which will be evaluated as a natural analog to secondary precipitation of spent nuclear fuel radionuclides. Geothermal areas, which provide a means to exercise numerical models of coupled thermal, hydrologic, and chemical processes, will be included. Colloidal transport of radionuclides at analog sites, environmental analog sites for future climate effects, and man-made analogs for ceramic materials that may find application in the engineered barrier system will also be included. Planned technical work to assess natural and man-made analogs is summarized in Sections 3.1.1.6, 3.1.2.2, and 3.1.5.

2.2.6.5 Summary of Technical Work to Address Element 5: A Performance Confirmation Plan

A performance confirmation plan has been developed (CRWMS M&O 1997d) that describes, in general terms, long-term testing and monitoring to confirm the assessment of principal factors affecting postclosure performance. The current plan is preliminary and will be revised once the design for the LA has been chosen. The revised performance confirmation plan will define the activities necessary to implement the requirements of 10 CFR Part 60, Subpart F, for the LA reference design. It will specify monitoring, testing, and analysis activities to be conducted to provide additional assurance that the postclosure performance objectives will be met.

The performance confirmation program defined in the revised 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 which is available for incorporation in the LA. Many of these testing and monitoring activities are ongoing components of the site characterization program. The parameters and concepts identified for performance confirmation will be based on models for natural barrier and engineered barrier performance that are available at the time of submittal of the LA. The plan will describe the process of integrating the additional information obtained after submittal of the LA into these models. Activities which are planned to extend to submittal of the LA and beyond, and which may therefore contribute to performance confirmation and be included in the revised plan, are noted in the summary tables which appear in Sections 3.1 and 3.2.

2.3 RATIONALE FOR TECHNICAL WORK NEEDED TO COMPLETE THE PRECLOSURE RADIOLOGICAL SAFETY CASE

Preclosure radiological safety criteria required for structures, systems, and components will be based on regulatory requirements, preclosure design basis events, and engineered safety features of the surface and subsurface facilities. The following sections summarize the preclosure radiological safety case process, the regulatory requirements, the technical work that has been completed, and the remaining technical work required for the site recommendation and the LA.

2.3.1 Preclosure Radiological Safety Case

DOE will develop a preclosure radiological safety case for the site recommendation and the LA that will demonstrate compliance with NRC regulatory requirements for the performance of the geologic repository during preclosure operations. Elements of this case will be based on a comprehensive safety analysis that identifies facility operations and waste emplacement scenarios from which design basis events; safety classification of structures, systems, and components; and system safety criteria are developed. As presented in this section, the strategy for developing this safety case enables the development of the repository system design and subsystem performance requirements to focus on those high priority features that are important to radiological safety. The preclosure radiological safety case will include an evaluation of the consequences of radiological exposure both at the site boundary and to occupational workers for the examined design basis events as mandated by 10 CFR 60.

2.3.1.1 Development of the Preclosure Radiological Safety Case

Preclosure radiological safety will be based on meeting radiological dose limits set by NRC regulatory requirements for offsite and the preclosure controlled area. To concentrate the design effort on those elements important to preclosure radiological safety, a strategy for design prioritization was developed to rank repository structures, systems, and components consistently. This prioritization effort is described in detail in Volume 2, Section 2.3.

Preclosure radiological safety will be provided by a combination of prevention and mitigation. Prevention is the use of features to ensure that the frequency of occurrence is minimized or eliminated. Mitigation is the use of features to ensure that public and worker exposures are within 10 CFR 60 and 10 CFR 20 limits for a given design basis event. The risk (combination of frequency and consequence) of a radiological release can be reduced by decreasing the occurrence frequency of an event sequence (i.e., prevention) and/or by decreasing the consequences of the event sequence (i.e., mitigation). Some design and operating features provide measures of both prevention and mitigation to reduce the risk of an event.

The preclosure radiological safety strategy for each operational function will include primary safety and defense in depth features. Primary radiological safety features are those that are determined by design basis event analyses to meet 10 CFR 60 and 10 CFR 20 requirements. Defense in depth features are those that are present or included to provide additional safety margins to reduce the overall safety risk.

The preclosure radiological safety case will allocate event and consequence prevention and mitigation functions to the structures, systems, and components based on their operational functions. The primary safety features (either prevention or mitigation) to satisfy the preclosure radiological safety requirements will be augmented by diverse and independent defense in depth features. The selection and reliance upon primary and defense in depth safety features will be supported by the design basis event analyses.

2.3.1.2 Systematic Identification of Design Basis Events

Importance to radiological safety is defined in 10 CFR 60 in terms of structures, systems, and components needed to show radiological dose compliance for potential design basis events. This definition is important in developing an acceptable preclosure safety case. There are two design basis event categories defined in 10 CFR 60. Category 1 describes "those natural and human induced events that are reasonably likely to occur regularly, moderately frequently, or one or more times before permanent closure of the geologic repository operations area." Category 2 consists of "other natural and human induced events that are considered unlikely, but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety."

Categorization of design basis events applies frequency screening to sequences of events as in the NRC interpretation of design basis events described in an NRC regulation (10 CFR 60.2, Design Basis Events). This process starts with a hazards assessment that identifies initiating events. Event-tree modeling is applied to define potential accident sequences (or scenarios) that could result in release of radionuclides to the environment. This event-tree modeling describes an accident sequence as an initiating event followed by one or more events that propagate, or fail to mitigate, the event sequence so that radionuclides are released. The frequency for such sequences is the product of the frequency of the initiating event times the conditional probability or probabilities of occurrence of each subsequent event in the sequence. Where appropriate, fault-tree modeling is applied to help estimate the frequency or probability of an event in the accident sequence.

Category 1 radiation dose limits to the public are defined in 10 CFR 60 by referencing 10 CFR 20. The total effective dose equivalent of 0.1 rem per year and 0.002 rem per hour to the offsite general public, and of 5 rem per year to repository workers from licensed operation, is the driving constraint for developing the preclosure radiological safety case. Radiological exposures for Category 1 design basis events are to be maintained as low as reasonably achievable.

As required during preclosure operations by 10 CFR 60 for Category 1 events, radiation exposures, radiation levels, and radioactive materials releases to unrestricted areas will be maintained within the limits specified by 10 CFR 20 and other generally applicable environmental standards for radioactivity established by EPA.

A primary constraint for developing the preclosure radiological safety approach is to design the operations area to prevent the general public from receiving more than a total effective dose equivalent of 5 rem per event for Category 2 events. The Category 2 radiological dose limits do not have 10 CFR 60 occupational dose limits.

Because different dose limits apply to each category, the quantitative frequencies of the design basis events are assessed and grouped into appropriate 10 CFR 60 categories. The threshold for designating a design basis event as credible is one occurrence in 1 million years. Based on the definitions from 10 CFR 60, categories are defined as follows:

Category	**Event Frequency**
Category 1	Sequence Frequency $\geq 1 \times 10^{-2}$ per year
Category 2	$1 \times 10^{-6} \leq$ Sequence Frequency $< 1 \times 10^{-2}$ per year
Not Credible (Beyond Design Basis)	Sequence Frequency $< 1 \times 10^{-6}$ per year

The 10 CFR 60 definition of Category 1 design basis events states that they occur one or more times before permanent closure of the geologic repository operations area. The Category 1 frequency is currently based on an operational period of 100 years. If preclosure subsurface monitoring operations are extended in the future, the event frequency of Category 1 events will be reexamined with respect to design basis event occurrence.

These frequency limits are interpreted as being applicable to any design basis event scenario that includes the initiating event and any subsequent failure resulting in a radionuclide release.

Additional safety assessments will be required for criticality control. Systems for processing, transporting, handling, storing, retrieving, emplacing, and containing radioactive waste are to be designed to ensure that nuclear criticality is not possible unless at least two unlikely, independent, and concurrent changes occur in the conditions essential to nuclear criticality safety per 10 CFR 60. Each system in the Monitored Geologic Repository will be designed for criticality safety assuming that a design basis event will occur. This control is achieved by showing that the calculated effective neutron multiplication factor is sufficiently below unity to provide at least a 5 percent margin, after allowances for calculational bias and experimental uncertainties.

Hazard and risk analyses have identified events with potential radiological consequences that are applicable to the repository system during preclosure. The methods used systematically identify and group design basis events according to 10 CFR 60 categories. The repository design will be analyzed for changes in design basis event frequency or consequence as greater design detail is developed. Design basis event identification is an iterative process and is coupled with requirements and design development.

Consequence analyses will determine whether the calculated dose rates are within applicable limits. If the calculated dose rates exceed applicable limits, the structures, systems, and components important to radiological safety are designated; new requirements are allocated; assumptions are revised; design configuration is revised (if necessary); and dose rates are recalculated and compared to applicable limits.

The evaluation of each design basis event will be prioritized and scheduled to support the site recommendation and the LA in accordance with the design prioritization described in Volume 2, Section 2.3.

2.3.1.3 Safety Classification of Structures, Systems, and Components Important to Radiological Safety

A classification analysis for the repository system has been completed (CRWMS M&O 1997a). This analysis determined what structures, systems, and components in the repository are quality affecting based on their radiological safety related functions. The analysis serves as the basis for the YMP *Q-List* (DOE 1998c). The *Q-List* is aligned with current repository system architecture and provides initial classification requirements and guidance to the design efforts. The classification analysis will be augmented by identifying and analyzing the design basis events to support a graded quality assurance classification of structures, systems, and components important to radiological safety before the site recommendation and the LA are submitted.

A graded approach to quality assurance will be used as primary and defense in depth radiological safety features are determined, because the features follow a hierarchy based on importance. For example, if the primary radiological safety feature provides adequate assurance that a possible design basis event is not credible, or that it meets regulatory requirements, the defense in depth features could invoke lower levels of graded approaches to safety margin and quality assurance controls. In such a case, the pedigree and rigor applied to the design and procurement of these defense in depth features may defer to good engineering practice, or only contain aspects of these features (i.e., maintenance or inspection) that may be elevated in safety significance to the degree dictated by the design basis event analysis.

2.3.1.4 Verification of System Design for Compliance to Requirements

Safety analyses will ensure compliance with 10 CFR 60. The safety functional requirements of the structures, systems, and components that design basis event analyses identify as important to radiological safety are included in the design requirements section of the system description documents. Quality assurance procedures require design reviews to ensure the design requirements were correctly selected and incorporated in the design. Qualification testing and/or evaluation will be performed in accordance with quality assurance procedures to verify that the design is in compliance with the design requirements.

NRC is currently revising regulations applicable to a repository at Yucca Mountain. If revisions to the regulations indicate that more safety analyses are necessary, the work will be performed and factored into the system description documents as appropriate. The design will then be verified and updated as necessary to ensure compliance with NRC regulations.

2.3.2 Use of Demonstrated Technology and Accepted Design Criteria

The Monitored Geologic Repository will be the first facility licensed under 10 CFR 60, but certain aspects of facilities already licensed under 10 CFR 50 and 10 CFR 72 may be applicable. The preclosure radiological safety approach and methods to demonstrate the safety of preclosure operations and demonstrate technical feasibility in compliance with 10 CFR 60 will require NRC review and acceptance.

DOE has established a formal compliance program to assess applicability and ensure compliance with applicable NRC regulatory guidance documents and related industry codes and standards. This assessment, along with supporting rationale, is formally documented as input to design criteria via compliance program guidance packages.

Compliance program guidance also is used as input for future activities and, where applicable, to engineering design guides and design analyses related to design basis events. The compliance program guidance used in the design processes is described in Volume 2, Section 2.

Compliance program guidance also applies to nonengineering topical areas considered in developing design criteria for engineered systems. The compliance program guidance typically will support developing acceptance criteria in the technical guidance document for nonengineering topical areas, such as:

- Site characteristics
- Emergency planning and preparedness
- Performance assessment
- Safeguards and security (including special nuclear material accountability control)
- Concept of operations and administrative controls
- Training and qualification
- Records management

2.3.3 Information Needs for Site Recommendation and License Application

Identification and evaluation of design basis events and their radiological consequences during preclosure operations are needed for the site recommendation and the LA to demonstrate that the preclosure radiological safety case for the repository system meets 10 CFR 60 requirements and applicable EPA standards. Radiological safety analysis (i.e., hazard analyses, design basis events analyses, radiological consequence analyses, and nuclear criticality analyses) comprise an ongoing effort to determine these consequences. Information needs to enable completion of these analyses include the following:

- Design and description of operations for receipt, handling, and emplacement of waste forms, including types of hardware, software, and human actions involved (alternative concepts should be identified in sufficient detail to permit analyses that can distinguish safety implications)
- Design basis drop capability for all containers of waste forms
- Physical characteristics of all waste forms to be received, including internal pressures, particulate quantities, and isotopic content
- Radionuclide source terms for each waste form

- Release fractions for all waste forms under accident conditions
- Chemical reactivity and pyrophoric behavior of all waste forms
- Characteristics and frequency of subsurface rockfall events
- Particulate retention factors for waste form containers
- Particulate retention factors for breached fuel rods
- Failure modes of structures, systems, and components used in receipt, handling, and emplacement of waste forms
- Meteorological models for the Yucca Mountain site

Analyses that will be completed before submittal of the site recommendation and the LA include the assessment of credible internal (e.g., drops) and external (e.g., earthquake and rainfall) events that will result in a final list of design basis events which the repository must be able to prevent or mitigate. Volume 2, Section 2.2, contains a discussion of safety analysis work that has been completed and the remaining analyses to be performed.

The classification of structures, systems, and components and the associated *Q-List* (DOE 1998c) is needed to ensure the LA design is produced in accordance with quality assurance procedures. Development of the *Q-List* is an iterative process and the revised information will be included in the site recommendation and the LA.

2.4 RATIONALE FOR ADDITIONAL TECHNICAL WORK TO SUPPORT DESIGN DECISIONS

This section provides the rationale for additional technical work needed to support the design decisions described in Section 2.2. A major thrust of the technical work remaining is to conduct the evaluations, including comparisons of options and alternatives, that will lead to selecting the reference design proposed to support the site recommendation and the LA. Selecting the design involves a sequence of decisions, some of which will require additional technical work. This section outlines the technical work to support the decisions regarding the following:

- Criticality issues
- Approaches to repository sealing and closure
- Evaluation of design alternatives

2.4.1 Technical Work to Address Criticality Issues

Current regulations require that nuclear criticality be prevented or its potential minimized in all phases of repository operation and after permanent closure. Two important aspects of the criticality analysis are securing NRC approval for using burnup credit to demonstrate criticality control, as described in Section 5.1.3 of Volume 2, and evaluating the effectiveness of measures to control criticality.

Technical work to be completed regarding nuclear criticality for the LA includes:

- Completing work to improve methods including determination of the range of parameters to describe the potentially critical configurations, refinement of the method for generating probability distributions of parameters affecting criticality, and estimation of consequences of potential criticalities.

- Completing work to justify the use of burnup credit and estimate the amount. Though the techniques being developed rest on scientific fact and well-understood theory, they do go beyond current practice in NRC licensing proceedings. Technical work is therefore needed to provide the case for the LA.

2.4.2 Technical Work to Address Approaches to Repository Sealing and Closure

The technical and environmental aspects of permanently closing the repository will be the subject of an NRC licensing proceeding related to a license amendment for permanent closure and decommissioning. However, the general approach will be evaluated before the LA is submitted to ensure that designs, construction, waste emplacement, and other operations do not preclude permanently closing the repository. Technical work will therefore include evaluating alternative approaches and closure activities including the possibility of a longer monitoring period and possible attendant maintenance needs.

The work will address sealing all underground openings, decommissioning surface facilities, reclaiming the site, establishing institutional barriers, and planning for postclosure performance monitoring. The work will consider the timing and materials to be used in sealing and closure and the integration of this information with other design information (e.g., the choice and emplacement of engineered barriers such as backfill).

Information needs for the work to be conducted before submitting the LA include:

- Storage and blending requirements
- Materials transport
- Backfill emplacement
- Structural components
- Impacts of schedule for closure on ventilation and underground facility maintenance
- Constraints on thermal loading requirements
- Impacts of schedule for closure on underground features and ground support
- Impact of schedule for closure on retrievability and backfilling

2.4.3 Technical Work to Evaluate Design Alternatives

A comprehensive and systematic design process will be conducted that will lead to selecting the design that will be the basis for the site recommendation and LA decisions. The design process will consider the potential advantages of alternative design features and design concepts.

The reference design was originally developed in 1996 to provide a consistent basis for making and comparing performance assessment evaluations. Its basis was the *Mined Geologic Disposal System Advanced Conceptual Design Report* (CRWMS M&O 1996), which assembled information contained in a variety of design reports and documents prepared for the most part in 1994 and 1995.

Substantial new or modified site information became available in late 1995 and early 1996. In addition, more focused and rigorous performance evaluations were being produced. Data from these sources resulted in more emphasis being placed on the engineered barrier system, which includes the waste package, the basic subsurface construction, and a potential set of other improvements. From that potential set, a short list of three items with the potential to enhance performance was selected as design options. These items are backfill, drip shield, and ceramic coating.

As site data and performance evaluations have provided more definitive information, expectations for the design element have been broadened in response to questions raised both within the current program and by outside reviewers. A design that meets an established performance standard is certainly an adequate solution. However, considerations of treatment of uncertainty, defense in depth, and margin can lead to questions about the likelihood of success in a licensing arena. The following are some of the additional considerations to be addressed:

- Are there fundamentally different (alternative) repository design concepts that could meet performance standards more effectively than the reference design?
- Are there design features that could be added or incorporated into either the reference design or any alternative concept(s) with significant benefit?

- Are there alternative concepts or features that, in addition to meeting performance standards, could provide additional advantages with regard to operational and regulatory and issues?

Substantial work has been performed previously that addresses, in part, these questions. However, that work was generally performed to respond to then current individual topical issues. Moreover, it was necessarily based on site, performance, materials and other data available at that time. That work still has value, but will now be reviewed and correlated with current information.

Current (fiscal year 1998) and planned (fiscal year 1999) work has been reviewed to address these questions, and do so in a way that provides comprehensive, sitewide and system-wide answers. As a result, and where feasible, some fiscal year 1998 work was redirected to support planning and beginning the comprehensive evaluation to select an initial LA design by May 1999. Updated and improved site models and data, new performance assessment evaluations, new engineering analyses, and new cost estimates will be prepared as needed to provide answers to the questions.

The specific approach to the evaluation will be developed as part of the process to establish the detailed evaluation criteria. This selection will consider the evaluation of possible design solutions against evaluation criteria related to postclosure performance, preclosure safety, licensing (e.g., transparency of the safety case), cost, schedule, flexibility of design, potential for environmental impacts, and potential for interference between site characterization activities and future construction.

Information needed to support evaluating these alternatives is specified in Section 8 of Volume 2. Much of this information will be provided from the work identified in Sections 2.2 and 2.3 of this volume. Other information will be obtained from reviews of past system studies. The remaining information needs will be met as a consequence of normal design considerations summarized in Section 3.

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3. TECHNICAL WORK PLANS

Section 3 is a comprehensive summary of the technical work planned and scheduled for the site recommendation in 2001 and the LA submittal in 2002. The work described in this section is divided into three areas—site investigations, design, and performance assessment—and is organized to coincide with the DOE Yucca Mountain project planning system. Together with Section 6, Costs, and Section 7, Schedules, this section presents an integrated plan for executing the required work. The schedule and the cost information are derived from the DOE project summary schedule with activities tied to DOE key milestones.

Work scope descriptions provided are cross-referenced to other parts of Volume 4. For example, work scope descriptions are linked back to the individual elements of the postclosure and preclosure safety cases described in Section 2. In addition, the individual elements of the work scope within site investigations, design, and performance are tied to the activities and milestones in Section 7. Further, cost information in Section 6 is also provided for the activities and milestones in Section 7. This arrangement thus presents an integrated picture of scope, schedule, and costs for the work to support submitting the LA. As discussed in Section 1.7.3 of Volume 1, the iteration of site characterization testing, design, and TSPA may change the priorities and plans presented in this section.

Section 3 also addresses how the work planned relates to the NRC key technical issues described in Section 4.3.3. As indicated in Section 4, technical work will not end on the day the LA is submitted to NRC. In addition to performance confirmation activities, some of the technical work will continue after the LA has been submitted. An obvious example is the drift-scale heater test (Section 3.1.4.3), which is planned to continue long after 2002. Other technical work, perhaps not yet foreseen, may also be needed to address information needs of NRC and other oversight organizations, or to resolve technical issues raised by data obtained and insights developed after 2002.

3.1 SITE INVESTIGATIONS

Site investigations have been planned to acquire the additional information needed to assemble the postclosure and preclosure safety cases, as discussed in Section 2. Planned work will supplement current information, to support the lines of evidence that comprise the safety cases. In this process the safety cases will also address the key technical issues identified by NRC.

Major goals of the site investigations are to reduce the remaining uncertainties for key principal factors affecting long-term performance of the repository and to provide necessary information for designing the repository to operate safely during the preclosure period. Work will also address multiple barrier performance through analysis of defense in depth, evaluate the impact of disruptive events, and use natural and man-made analogs to provide additional confidence in performance assessment.

Plans for additional site investigation work are grouped according to seven general technical themes as follows:

- Geologic framework and disruptive events
- Unsaturated zone processes
- Saturated zone processes
- Thermal testing
- Near-field environment and coupled processes
- Performance confirmation
- Management and integration, report preparation and review, and field support

The first group, geologic framework, which includes the distribution and properties of geologic units and structures, forms the basis for modeling water movement and the transport of radionuclides. Disruptive events (e.g., earthquakes and volcanoes) are included in this group because geology plays a large role in understanding the likelihood of these events. The next two groups, unsaturated and saturated zone processes, focus on the movement of water and the transport of radionuclides that might be released from a repository. This work will provide additional support for the two flow and transport process models used to assess long-term performance. The next two groups of plans are also related, and the work will investigate how a repository could perturb the existing environment, especially from the effects of waste-generated heat. Results of this work will support models of the geomechanical, geohydrological, and geochemical processes that will be affected by repository materials and thermal effects. The sixth group, performance confirmation, includes work to monitor the behavior of a repository as it is built and operated. This information will help to confirm whether or not observed behavior conforms to the range of behavior predicted by models used for the licensing process. The last group represents management and support work, such as qualifying data and models that support safety lines of evidence.

Within these groups, work is further divided into work categories. For each work category, the text and tables identify milestones from the project, the schedule (Section 7), and applicable activity numbers from the project summary schedule for which cost estimates are provided (Section 6).

3.1.1 Geologic Framework and Disruptive Events Group

The characterization of geologic features and processes at Yucca Mountain is essentially complete. Information on the distribution of rock units and the offsetting faults has been incorporated into a model of the geologic framework for the site. This model has been combined with spatial models of rock properties and mineral distribution to produce the three-dimensional integrated site model that is used for performance assessment to model how water moves through the mountain.

Remaining work will focus on completing the geologic framework for the saturated zone model. Tasks will include mapping surface geology and new underground excavations, testing samples from excavations and boreholes, incorporating new data into geologic databases and models, and providing geologic support to performance assessment and design activities.

Studies of disruptive events have also reached a level of maturity for which only limited additional information is required. Hazard assessments show the likelihood of volcanic eruptions or fault displacement through the potential repository is low. The hazard of ground motion from earthquakes has been characterized and has been factored into design and performance assessment activities. Additional work will ascertain the site-specific properties of rock and soil, particularly at the potential location of repository surface facilities. These data will be used to develop the seismic designs for facilities important to safety. New findings by investigators from outside the project will also be reviewed to determine if there are any implications for current conclusions. If necessary, new technical work will be done.

Safety Case. Work plans in this area primarily address the first and third elements of the postclosure safety case, which are assessment of expected postclosure performance and consideration of disruptive processes and events. The geologic framework model supports the assessment of postclosure performance by providing the underlying basis for the models of water flow, and radionuclide transport and near-field environment processes. Remaining work to complete the characterization of disruptive processes will focus on preclosure seismic issues, and thus, will also support the preclosure safety case.

NRC Key Technical Issues. This work will also address aspects of eight NRC key technical issues. Determining geologic conditions in the cross drift and testing cross drift samples will address Repository Design and Thermal-Mechanical Effects (Section 4.3.3.6). Finalizing seismic design inputs addresses these issues as well as aspects of Structural Deformation and Seismicity (Section 4.3.3.2). In addition, the geologic framework model provides information pertaining to Thermal Effects on Flow (Section 4.3.3.5), TSPA and Integration (Section 4.3.3.7), Unsaturated and Saturated Flow Under Isothermal Conditions (Section 4.3.3.9), and Radionuclide Transport (Section 4.3.3.10). Finally, considering disruptive events addresses aspects of Structural Deformation and Seismicity (Section 4.3.3.2), and Igneous Activity (Section 4.3.3.1), and the subissue concerning disruptive events associated with activities related to Development of the U.S. Environmental Protection Agency (EPA) Yucca Mountain Standard (Section 4.3.3.8).

Project Summary Schedule Activities. The tasks for each work category in the following sections will be completed to support the site recommendation and the LA. Each work category corresponds to one or more project summary schedule activities, which in turn often cover more than one work category. The project summary schedule activities that include work for the geologic framework and disruptive events are 2050, Testing for Enhanced Characterization of the Repository Block; 2210, Develop Hydrologic Framework/Evaluate Disruptive Events for SR; and 8601, Confirm Hydrologic Framework/Evaluate Disruptive Events for Construction Authorization. Though work for Activity 2050 will cut across many technical areas, only the geologic component is discussed in this section.

The planned tasks for geologic framework and disruptive events will include the following:

- Complete the geologic and geotechnical mapping of subsurface excavations
- Complete the surface geologic mapping that supports developing the model of how water moves through the saturated zone below the water table
- Update and maintain the three-dimensional geologic and integrated site model
- Finalize seismic engineering design input
- Provide geological and geophysical support to other ongoing activities
- Continue convergence monitoring of the Exploratory Studies Facility and cross drift excavations
- Test rock properties for new boreholes and excavations

These tasks are described in more detail in the following sections. The applicability of each work category to the postclosure and preclosure safety cases is summarized in Table 3-1. The "Priority to SR/LA" assessed in Section 2.2 is also shown on the table as a number from "0" to "3" (lowest to highest priority). The table lists the project summary schedule activities associated with each work category, for which cost and schedule information is given in Sections 6 and 7.

3.1.1.1 Geologic and Geotechnical Mapping of Subsurface Excavations

Geological and geotechnical logs will be collected to document the rock types and geologic structure of excavations, which are underway to provide additional testing facilities underground. These data will be incorporated into the three-dimensional geologic model, and form part of the framework within which test results are interpreted. These efforts will therefore support field testing to address the principal factors for drift seepage, and the effects of heat and excavation on flow, which are assigned relatively high priority in Section 2.2 (Table 2-2). Geologic and geotechnical data collection will also help in understanding rock conditions that will be considered in repository design. Planned work includes the following:

- Complete the *Cross-Drift Geologic and Geotechnical Report* to support Milestones M2MQ, Select Initial SR/LA Design and Options; M2HC, Decide Unsaturated Zone Model for SR/LA; and M2HE, Decide Unsaturated Zone Transport Models for SR/LA.
- Map alcoves and niches constructed in the Exploratory Studies Facility and cross drift for seepage and percolation studies to support Milestones M2HC and M2HE.

Table 3-1. Application of Planned Technical Work on the Geologic Framework and Disruptive Events to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Postclosure Safety Case[1]	Elements of the Postclosure Safety Case[1]: 1. Expected Postclosure Performance: Principal Factors Supported by Technical Work	Priority to SR/LA[2]	2. Design Margin/ Defense in Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Performance Confirmation	Preclosure Safety Case[3]	PSS Activities[4]
Geologic and Geotechnical Mapping of Subsurface Excavations (3.1.1.1)	√	Flow and transport in the unsaturated zone	3						2050 8601
Surface Geological Mapping (3.1.1.2)	√	Flow and transport in the saturated zone	1						2210 8601
Three-Dimensional Geologic and Integrated Site Model (3.1.1.3)	√	Flow and transport in the unsaturated zone Flow and transport in the saturated zone	3 1						2210 8601
Seismic Engineering Design Input (3.1.1.4)					√			√	2210 7027
Geophysical Investigations (3.1.1.5)	√	Percolation to depth Flow and transport in the unsaturated zone	2 3						2050 8601 2215
Geological/ Geophysical Support to Other Activities (3.1.1.6)	√	Effects of heat and excavation on flow Dripping onto waste packages Flow and transport in the unsaturated zone Flow and transport in the saturated zone	2 2 3 1			√	√		2210 8601 2025 2029
In Situ Design Confirmation (3.1.1.7)	√	Effects of heat and excavation on flow	2				√		2210
Thermal-Mechanical Rock Properties Testing (3.1.1.8)	√	Effects of heat and excavation on flow	2						2050 2210

[1] Postclosure safety case described in Section 2.2.

[2] Postclosure work prioritization for site recommendation and LA (SR/LA) described in Section 2.2.4.

[3] Preclosure safety case described in Section 2.3.

[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

3.1.1.2 Surface Geologic Mapping

Surface geologic mapping at a scale of 1:50,000 will be completed to support the saturated zone flow model (Milestones M2HD and M2HF, in Section 7). Additional geologic mapping is an important step in refining models for radionuclide transport in the saturated zone. Further work on transport in the saturated zone has been assigned a relatively low priority in Section 2.2 (Table 2-2), so the extent of this work will be limited. The mapping effort provides the geologic framework to define the hydrogeologic flow units and fault structures to be included in the model.

3.1.1.3 Three-Dimensional Geologic and Integrated Site Model

The three-dimensional geologic and integrated site model is the framework on which the unsaturated and saturated flow and transport models are developed (Milestones M2HC, M2HD, M2HE, and M2HF). These process models, in turn, support the assessment of postclosure repository performance. The process model for flow and transport in the unsaturated zone has been assigned a relatively high priority for the LA, in Section 2.2 (Table 2-2). The existing geologic model, including rock properties and mineralogical components, will be updated with new information from the cross drift, the USW WT-24 and USW SD-6 boreholes, surface facility boreholes, and other data as they become available. The updated geologic model will support the process models that are used to assess repository performance.

3.1.1.4 Seismic Engineering Design Input

Ground motion input for seismic design of the repository will be finalized and the preliminary report *Seismic Design Basis Inputs for a High-Level Radioactive Waste Repository at Yucca Mountain* will be revised to support design Milestone M2MQ. Planned activities will include the following:

- Acquire site-specific soil and rock properties as part of foundation studies for the Waste Handling Building. Investigations will include drilling of a shallow borehole into bedrock and measuring shear wave velocity.
- Determine site-specific values for near-surface attenuation of ground motion.
- Continue to monitor earthquakes at the surface and in boreholes near the repository and surface facilities, to better understand the attenuation of ground motion and its spatial variability.

These tasks will provide additional support to assessing disruptive processes and events for the postclosure safety case, and developing inputs for the seismic design of surface facilities important to safety.

3.1.1.5 Geophysical Investigations

Assessment by seismic tomographic methods will be completed using data collected in the Exploratory Studies Facility and the cross drift to evaluate the quality of subsurface rock and define zones of more fractured rock. The data will provide a non-invasive, three-dimensional view of rock quality across the repository block where there are no boreholes or drifts, and thus will support design Milestones M2MQ and M2MV (Section 7). The results will also support modeling water percolation through Yucca Mountain (Milestone M2HC) and the transport of radionuclides in the unsaturated zone below the repository horizon (Milestone M2HE). These principal factors are assigned relatively high priorities of "2" and "3," respectively, in Section 2.2 (Table 2-2).

3.1.1.6 Geological and Geophysical Support to Other Activities

Geological and geophysical support will be provided to other ongoing activities. These efforts will include the following:

- Interpret and review geologic contacts and geophysical logs, including those obtained for boreholes drilled by Nye County, to support refining models for flow and transport in the saturated zone.
- Provide geologic cross sections for flow models. This work will support developing refined models for flow and transport in the unsaturated zone, which is assigned relatively high priority for the LA, in Section 2.2 (Table 2-2). Modeling saturated zone flow and transport, a lower priority for the LA, will also be supported.
- Support geologic characterization of the thermal test area in the Exploratory Studies Facility and the test area at Busted Butte. These tests also support models for flow and transport in the unsaturated zone, effects of heat and excavation on flow, and other principal factors for postclosure performance.

- Provide a geologic evaluation of natural analog sites and conditions. This effort will support the postclosure safety case, which will use natural analogs to help mitigate uncertainty in assessments of system performance.

- Provide geologic input for designing performance confirmation tests to prepare for the LA.

3.1.1.7 In Situ Design Confirmation (Convergence Monitoring)

To increase confidence in the performance of underground excavations at Yucca Mountain, convergence monitoring will continue in the Exploratory Studies Facility and the cross drift. These measurements indicate how the ground support system interacts with the surrounding rock mass, and will be used primarily as design input for repository ground support systems and to support performance confirmation. The planned work will support LA design Milestones M2MQ and M2MV (Section 7).

3.1.1.8 Thermal-Mechanical Rock Properties Testing

Laboratory and in situ testing will be done to determine the thermal and mechanical properties of the host rock at Yucca Mountain. Borehole jack tests and in situ stress tests will be performed in the cross drift. The in situ tests will be complemented by laboratory thermal and mechanical tests on samples from the cross drift and from Borehole SD-6. Empirical models of strength, modulus, and thermal properties of the rock mass will be updated. The planned work will support LA design Milestones M2MQ and M2MV and near-field models Milestone M2JC (Section 7).

3.1.2 Unsaturated Zone Processes Group

This group of work plans will address processes operating from the land surface to the water table. This zone is called the unsaturated zone and at Yucca Mountain it consists of the soils at the land surface, and volcanic rocks beneath. Modeling and experimental studies of unsaturated zone processes will continue, and additional field tests will be undertaken, to support the site recommendation and the LA. These studies are expected to significantly reduce uncertainty as to how water flows through the unsaturated zone and interacts with drift openings, and how that water could transport radionuclides to the water table.

Key topics considered in studying unsaturated zone processes are future climate, precipitation, infiltration of water at the surface, percolation of water through the fine pores and fractures in the rock, and seepage into drifts. Flow generally proceeds downward along fractures, and the spatial distribution is known generally, but not in detail. Some of this flow is expected to be sporadic, reflecting the episodic nature of storms at the surface. At other locations or times, the flow is expected to be more uniform resulting from mediation of the episodic infiltration as it flows through surface soils and multiple rock layers.

Two important processes that retard the transport of radionuclides through the unsaturated zone are matrix diffusion and sorption. Matrix diffusion involves the movement of groundwater, with dissolved radionuclides, from fractures in the rock to pores within the rock matrix. This redistribution retards the movement of the dissolved radionuclides because water flows much more quickly through the fractures than through the matrix of the rock. Sorption is controlled by the mineralogy of the local rocks and fracture coating minerals, as well as the chemistry of the groundwater, which in turn affect the solubility and speciation (specific chemical form) of the dissolved radionuclides. The mechanism of sorption involves the attachment of dissolved radionuclides to fine particles in the groundwater (colloids) or minerals in the local rocks for varying lengths of time (minutes to thousands of years). This retention can occur as either a chemical bond, called absorption, or a physical surface attraction, called adsorption.

Planned work to refine models for unsaturated zone processes will focus on seepage into drifts, formation of radionuclide-bearing colloids, and transport of colloidal and dissolved radionuclides through the unsaturated zone. Planned work will monitor the rate of percolation flow through the host rock, and measure the conditions at which percolation causes seepage into drifts. Monitoring moisture migration in the Exploratory Studies Facility will contribute to understanding how repository ventilation could potentially dry out the host rock. The work will directly address principal factors for percolation, seepage into drifts, and unsaturated flow and transport, which are assigned relatively high priority for the LA, in Section 2.2 (Table 2-2).

Safety Case. The work in this category will primarily address the first element of the postclosure safety case (assessment of expected postclosure performance) by reducing uncertainties related to radionuclide transport and movement of water through Yucca Mountain. In addition, a portion of the effort will support the fourth element of the postclosure safety case by compiling information on natural and man-made analogs.

NRC Key Technical Issues. Activities to characterize the unsaturated zone will provide additional data to address several NRC key technical issues. Test results will directly address the key technical issue for Unsaturated and Saturated Flow Under Isothermal Conditions (Section 4.3.3.9). This key technical issue is concerned with all aspects of the ambient hydrogeologic regime at Yucca Mountain that have the potential to compromise repository performance. Additional testing will address the subissue concerning the amount and extent of infiltration at Yucca Mountain, and two subissues concerning the movement of groundwater in the unsaturated zone above the repository. Seepage is also related to the subissue concerning unsaturated zone percolation discussed in Section 4.3.3.9 and the subissue concerning effects of coupled processes on seepage discussed in Section 4.3.3.3.

Studies of radionuclide transport will also directly address the NRC key technical issue of Radionuclide Transport (Section 4.3.3.10), regarding four related subissues as follows:

- Identify radionuclides for which calculated dose rates are potentially unacceptable
- Evaluate how the geochemical and hydrological properties of the saturated and unsaturated zones control radionuclide transport
- Assess conceptual models and mathematical approaches to modeling the transport of radionuclides
- Determine the sensitivity of overall repository performance to parameters affecting radionuclide transport

The information will also address the key technical issue, TSPA and Integration (Section 4.3.3.7). This information from field testing will be used to verify process models that are used in TSPA, and to evaluate the ranges of model input parameters that will be used to represent the uncertainty of these models.

Project Summary Schedule Activities. The work described in categories in the following sections will be completed to support the site recommendation and the LA. Each work category corresponds with one or more project summary schedule activities in the project budget and schedule, and many project summary schedule activities cover more than one work category. The project summary schedule activities which include work for unsaturated zone processes are 2025, Data and Analysis to Address Seepage, Unsaturated Zone Flow and Transport for TSPA-SR; 2027, Modeling to Evaluate Seepage and Unsaturated Zone Flow and Transport for TSPA-SR; 2050, Testing for Enhanced Characterization of the Repository Block; 2215, Data and Analysis to Update Seepage and Unsaturated Zone Flow and Transport Models for SR/LA; 6105, Science Support to License Application; 8605, Data and Analysis to Confirm Results on Seepage, Unsaturated Zone Flow and Transport for Construction Authorization; and 8607, Modeling to Confirm Conclusions on Seepage and Unsaturated Zone Flow and Transport-Construction Authorization.

These work categories are described in more detail in the following sections. The applicability of each work category to the postclosure and preclosure safety cases is summarized in Table 3-2. The "Pri-

Table 3-2. Application of Planned Technical Work on Unsaturated Zone Processes to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Post-closure Safety Case[1]	Elements of the Postclosure Safety Case[1]: 1. Expected Postclosure Performance — Principal Factors Supported by Technical Work	Priority to SR/LA[2]	2. Design Margin/ Defense in Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Per-formance Confirmation	Pre-closure Safety Case[3]	PSS Activities[4]
Formation of Radionuclide-Bearing Colloids (3.1.2.1)	√	Formation and transport of radio-nuclide-bearing colloids	2			√			2025 2215 8605
		Transport through and out of engineered barrier system	1						
		Flow and transport in the unsaturated zone	3						
Transport in the Unsaturated Zone (3.1.2.2)	√	Formation and transport of radio-nuclide-bearing colloids	2			√			2025 2215 8605
		Flow and transport in the unsaturated zone	3						
Water Movement through the Unsat-urated Zone (3.1.2.3)	√	Precipitation and infiltration into the mountain	1						2025 2050 2215 8605
		Percolation to depth	2						
		Seepage into drifts	3						
		Flow and transport in the unsaturated zone	3						
Mountain-Scale Unsaturated Zone Flow and Transport Model (3.1.2.4)	√	Percolation to depth	2						2027 2050 8607
		Flow and transport in the unsaturated zone	3						
Drift-Scale Unsat-urated Zone Flow and Transport Model (3.1.2.5)	√	Seepage into drifts	3						2025 2050 2215 8605 8607
		Effects of heat and excavation on flow (drift scale)	2						
		Flow and transport in the unsaturated zone	3						

[1] Postclosure safety case described in Section 2.2.
[2] Postclosure work prioritization for site recommendation and LA (SR/LA) described in Section 2.2.4.
[3] Preclosure safety case described in Section 2.3.
[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

ority to SR/LA" assessed in Section 2.2 is also shown on the table as a number from "0" to "3" (lowest to highest priority). The table lists the project summary schedule activities associated with each work category, for which cost and schedule information is given in Sections 6 and 7.

3.1.2.1 Formation of Radionuclide-Bearing Colloids

Colloids, extremely small particles suspended in water, are important to understanding transport through the unsaturated zone because radionuclides may bind to them. Laboratory studies are planned to address uncertainties about the formation and stability of colloids and their interactions with radionuclides, particularly plutonium. The principal factor for colloid formation and transport is assigned a relatively high priority for the LA in Section 2.2 (Table 2-2). Emphasis on colloid studies in planned work is also consistent with the findings of TSPA (Volume 3, Section 6.4.14). Colloid work will specifically address NRC subissues (Section 4.3.3) that relate to the rate of colloid production by the degradation of spent nuclear fuel, the transport of colloids through the engineered and natural barriers, the radionuclides that are transported colloidally and may be significant to total system performance, and the operant geochemical and hydrological controls on colloidal transport. These results will support Milestones M2HE and M2HF (Section 7). Transport through the unsaturated zone via colloids will also be addressed through examination of analogs.

3.1.2.2 Transport in the Unsaturated Zone

Both field and laboratory tests will be used to reduce uncertainties related to the transport of radionuclides through the unsaturated zone. The field test at Busted Butte will continue to observe how radionuclides and colloids are transported in the Calico Hills nonwelded rock unit that is also present in the unsaturated zone under the repository horizon. The results of the field test will be used to update the conceptual understanding and parameters used in the existing unsaturated zone flow and transport model.

In addition, planned laboratory tests will be conducted to prepare for the LA as follows:

- Complete the study of the speciation, solubility, and sorption of the key radionuclides technetium, plutonium, and neptunium, which are shown by TSPA to comprise much of the inventory of radionuclides that may be transported to the biosphere (Volume 3).

- Provide data on radionuclide transport in alluvium, which will be part of the transport pathway at distances approaching 20 km (12 miles) from the site.

- Complete the documentation on all previous and current laboratory studies of radionuclide sorption, speciation, and transport, during 1999.

- Evaluate the transport of radionuclides at former DOE weapons production facilities for possible analogs to Yucca Mountain.

- Update the three-dimensional mineralogic and petrologic model of rock characteristics in the host rock and along transport pathways to the biosphere, to provide a more complete description of the spatial distribution of sorptive minerals in transport models.

Planned work will investigate natural and man-made analog sites, including Peña Blanca and the Nevada Test Site, respectively, to reduce uncertainties about long-term radionuclide transport. Analog data will support the postclosure safety case for the LA.

The planned work is expected to substantially increase confidence in assessments of unsaturated zone transport, particularly from the contributions of the Busted Butte field test and supporting studies. The principal factor for unsaturated zone transport is assigned a relatively high priority for the LA, in Section 2.2 (Table 2-2). These studies will support resolving the NRC key technical issue for Radionuclide Transport, and the subissues concerning identification of radionuclides that require retardation to meet performance requirements and evaluation of geochemical and hydrological controls on radionuclide transport (Section 4.3.3.10). The planned work will support Milestone M2HE, and some elements will also support Milestone M2HF (Section 7).

3.1.2.3 Water Movement through the Unsaturated Zone

Study of percolation and seepage at the drift scale will continue to provide insights about the amount of moisture that may contact waste packages. Planned work will further reduce uncertainties regarding percolation in geologic units that comprise most of the waste-emplacement horizon. This work will include the following:

- Excavate two niches and prepare two fracture-matrix test beds in the cross drift, to collect key data to quantify and assess the natural and engineered barriers inside the drifts. Seepage and fracture-matrix interaction tests will be performed in the lower Topopah Spring units (lithophysal and nonlithophysal) that comprise the potential repository horizon. Planned work will include liquid release tests and long-term tracer injection tests. The niche tests will be similar to previous tests in the Exploratory Studies Facility and will provide direct results on the magnitude and distribution of seepage at field-scale, in situ conditions. The fracture-matrix interaction tests will provide key data on the mixing of fracture water with matrix pore water and the associated potential for radionuclide retardation.

- Perform geochemical and isotopic analyses to determine where water has flowed in the repository host rock during the geologic past. These analyses will check for the presence and concentrations of chemical components such as chloride, bromide, and sulfate. The results will be used to identify fast paths and travel times. Geochemical and isotopic data provide a critical capability for field confirmation of the site-scale infiltration model and site-scale unsaturated zone flow and transport models. Planned studies include the following:

  - Analyze chloride, bromide, and chlorine-36 in samples collected from the cross drift, the Exploratory Studies Facility, and at the Busted Butte test site. These measurements are key indicators of flow pathways and infiltration rate. In addition, data from all parts of the repository block that have been explored by boreholes and underground excavation will be made available.

  - Develop an integrated interpretation of the geochemical and isotopic data for the unsaturated zone, in conjunction with mineralogic and structural data. Assess the consistency of geochemical data with existing flow and transport models.

  - Provide analytical support to other planned field tests for which construction water will contain tracers such as bromide.

  The additional planned work will support the principal factors for percolation and seepage into drifts, which are assigned relatively high priority for the LA, in Section 2.2 (Table 2-2).

- Perform a controlled study of percolation from the newly constructed cross drift to the underlying Exploratory Studies Facility main drift. The cross drift infiltration experiment in the crossover alcove will provide data on percolation rates through fractured welded tuffs under controlled boundary conditions, over flow path distances of up to 20 m (65 feet). Data on the hydraulic properties of fractures, fracture distributions, fracture apertures, fracture-matrix interactions, and matrix hydraulic properties will be collected and analyzed. The controlled boundary conditions and the flow regime measured during this experiment will provide field data needed for testing and calibrating unsaturated zone flow models. These data will be used to evaluate spatial variability and to determine bounds for the percolation flux at the repository horizon, the fraction of percolation flux that seeps into an excavated niche, and the diversion flow established around an excavated niche.

- Determine the age of minerals to provide information on the history of water movement through the site. The distribution of ages for minerals deposited along fracture walls relates to the history of water movement through Yucca Mountain. These measurements will indicate when and where water has moved through the host rock and will also reveal characteristics of the water, including chemical composition and temperature. The data will be incorporated in a three-dimensional model that describes the history of water movement through the unsaturated zone. The following studies are planned:

  - Complete the dating and isotopic analyses of fracture minerals already collected from the Exploratory Studies Facility

  - Extend studies of calcite and opal fracture filling minerals in the Exploratory Studies Facility main drift and its alcoves, to similar deposits sampled in the cross drift

  - Develop an integrated interpretation of all the available mineral isotopic and chemical data, to form a mineral-based conceptual model of water movement through the unsaturated zone

- Monitor moisture in the Exploratory Studies Facility and the cross drift. Moisture monitoring activities in Alcoves 1 and 7 of the Exploratory Studies Facility will continue and monitoring in the cross drift will be established to accomplish the following:

  - Determine the moisture balance within the underground openings.

  - Determine the effects of underground ventilation and the water used to control dust while the tunnel boring machine is operating. Measure the physical state of water in the rock surrounding the cross drift and evaluate how water applied during construction percolates away from the tunnel.

The planned tests will directly support the principal factors for percolation, seepage into drifts, and unsaturated zone flow and transport, which are assigned relatively high priority for the LA, in Section 2.2 (Table 2-2). The planned work will address subissues associated with three NRC key technical issues. Much of the work will relate to subissues concerning present and future percolation flux at and above the repository horizon (Section 4.3.3.9). Other work will address percolation and the potential for seepage into drifts, as well as the effects of coupled processes on seepage associated with Evolution of the Near-Field Environment (Section 4.3.3.3). The planned work will also address the NRC key technical issue for Thermal Effects on Flow (Section 4.3.3.5). The results from these tests will support Milestone M2HE (Section 7).

The data collected in these experimental activities will be modeled at both the mountain scale and the drift scale, as discussed in the following sections. Information from natural and man-made analogs will be used to build confidence in models of water movement through the unsaturated zone.

3.1.2.4 The Mountain-Scale Unsaturated Zone Flow and Transport Model

Movement of groundwater above the water table and transport of radionuclides below the repository horizon are simulated by the mountain-scale model of the unsaturated zone. The model is bounded by infiltration at the ground surface and the water table below. The unsaturated zone is represented by rock layers based on the geologic framework model for Yucca Mountain, and by laboratory measurements of physical and hydrologic properties of the rock matrix. Hydrogeologic properties are defined for each layer and calibrated by comparing model calculations to measurements of in situ rock saturation, chemistry of water collected at depth, and other observations. Current confidence in this model is limited because the interaction of fracture water with pore water in the rock matrix is uncertain, as are the processes and rock properties that control the transport of chemical species. Planned work to reduce these uncertainties will include the following:

- Update model parameters using results from transport studies at the Busted Butte field-test, testing in niches within the cross drift, thermal tests, and laboratory measurements.

- Improve the model by incorporating better representation of two geologic units; the Paintbrush tuff nonwelded unit, which lies above the repository horizon and diverts infiltration laterally and away from the potential waste emplacement area, and the Calico Hills nonwelded unit beneath the repository.

- Incorporate better representation of fault zones and percolation.

- Improve the manner in which spatial variability in hydrogeologic properties is represented.

Results from the mountain-scale model will be abstracted for use in the TSPA for the LA and for evaluation of model sensitivity and uncertainty. Model sensitivity analyses will include evaluating the transport of radionuclides after postulated early failure of one, or a small number of, separate waste packages.

These planned activities will directly address the principal factor for unsaturated flow and transport, which is assigned a relatively high priority for the LA, in Section 2.2 (Table 2-2). The results of mountain-scale flow and transport modeling will support Milestone M2HE (Section 7). Work will continue past the date of this milestone to test the sensitivity of model results to uncertainties in input parameters and to evaluate any new data. This work will provide added confidence to the information contained in the LA.

3.1.2.5 Drift-Scale Unsaturated Zone Flow and Transport Model

This model simulates seepage of percolation flow into drifts. It is used to represent the period after the repository has cooled to ambient temperature, from several thousand years after repository closure through the remainder of the postclosure period. During this period, water can enter the drifts and contact the waste packages without influence from thermal effects, except for long lasting changes in the host rock properties caused by coupled processes during the thermal period. For long-term performance, the quantity of seepage into drifts is important for calculating the degradation rates of the waste packages and waste forms. Reducing the remaining uncertainty in the drift-scale flow and transport model will be accomplished by performing the following:

- Recalibrate the model based on field test results

- Revise the calculations for flow along the drift wall surface

- Include the effects of a silica/calcite cap that may precipitate in the rock over the drift

- Include the effects of alternative drift geometry caused by rockfall

As for the mountain-scale model, results from the model will be abstracted for use in the TSPA for the LA and for evaluation of model sensitivity and uncertainty. Model sensitivity analyses will include evaluating transport of radionuclides after a postulated early failure of one, or a small number of, separate waste packages.

These planned activities will directly address the principal factor for unsaturated flow and transport, which is assigned a relatively high priority for the LA, in Section 2.2 (Table 2-2). The results of drift-scale flow and transport modeling will support Milestone M2HE (Section 7). Work will continue past the date of this milestone to test the sensitivity of model results to uncertainties in input parameters and to evaluate any new data. This work will provide added confidence to the information contained in the LA.

3.1.3 Saturated Zone Processes

This group of work plans will address the movement of water in the saturated zone, below the water table. Collecting data for the saturated zone will include monitoring changes in the depth of the water table, measuring the rate at which the water moves through the rock and fractures and measuring the distribution of water characteristics. These characteristics include apparent radiocarbon age, temperature, chemical and isotopic content, pH, and oxidation/reduction potential.

Results will be factored into models that represent the rock layers in the saturated zone, and the water flowing through them. The work will contribute to understanding the paths that water follows after it enters the saturated zone. It will also provide additional information about hydrogeologic features of the saturated zone, such as rock units and faults that enhance or hinder the movement of water. Mixing of the waters in different rock units comprising the saturated zone is important for assessing the long-term performance of a repository. Planned work will further develop models of saturated zone flow at site and regional scales.

In addition to representing the movement of water, the models also address the transport of radionuclides. As discussed for the unsaturated zone, various mechanisms can slow the transport of radionuclides relative to the movement of water carrying them. Within the saturated zone, the mechanisms of matrix diffusion and sorption are supplemented by dilution. Dilution results if the small volume of water percolating down through Yucca Mountain mixes with the larger volume of water present in the saturated zone, thereby reducing the concentrations of radionuclides.

Planned work will reduce uncertainties concerning flow and transport in the saturated zone. Work will focus on providing hydraulic parameters for the Prow Pass geologic unit, on collecting and analyzing data to interface regional and site-scale models, and on characterizing the geochemical conditions affecting transport of neptunium and other radionuclides important to long-term performance. The planned work will support the principal factor for saturated zone flow and transport, which is assigned a relatively low priority for the LA, in Section 2.2 (Table 2-2). As pointed out in Section 2.2, the scope of additional saturated zone characterization that can support the LA is limited because of the time available. The work planned for this category is planned accordingly.

Safety Case. Work in this category will address the first element of the postclosure safety case, assessment of expected postclosure performance. The results will be incorporated into improved models of saturated zone flow and transport that will support assessments of postclosure repository performance for the LA.

NRC Key Technical Issues. The additional work planned to characterize the saturated zone will address the NRC key technical issue for Unsaturated and Saturated Flow Under Isothermal Conditions (Section 4.3.3.9). Results will directly address the subissue concerning flow conditions in the saturated zone. Studies of radionuclide transport will directly address the NRC key technical issue for Radionuclide Transport (Section 4.3.3.10), especially the subissues related to hydrologic and geochemical controls on radionuclide transport and the potential for dilution. The models will also address the key technical issue of TSPA and Integration, especially the subissue concerning model abstraction and data integration (Section 4.3.3.7).

Project Summary Schedule Activities. The work categories in the following sections will be completed to support the site recommendation and the LA. Each work category corresponds with one or more project summary schedule activities in the project budget and schedule, and many project summary schedule activities cover more than one work category. The project summary schedule activities which include work for saturated zone processes are 2029, Data and Analysis to Evaluate Dilution along Pathways in the Saturated Zone for TSPA-SR; 2031, Modeling to Evaluate Dilution and Pathways in the Saturated Zone for TSPA-SR;

2245, Data and Analysis to Update Dilution along Pathways in the Saturated Zone for SR/LA; 6105, Science Support to License Application; 8609, Data and Analysis to Confirm Conclusions on Dilution along Pathways in the Saturated Zone for Construction Authorization; and 8611, Modeling to Confirm Conclusions on Dilution along Pathways in the Saturated Zone for Construction Authorization.

These work categories are described in more detail in the following sections. The applicability of each category to the postclosure and preclosure safety cases is summarized in Table 3-3. The "Priority to SR/LA" assessed in Section 2.2 is also shown on the table as a number from "0" to "3" (lowest to highest priority). The table lists the project summary schedule activities associated with each work category, for which cost and schedule information is given in Sections 6 and 7.

3.1.3.1 C-Wells Hydraulic Testing

Hydraulic testing at the C-wells complex of boreholes will be completed. The additional work will consist of conservative and reactive tracer tests, and companion aquifer hydraulic tests, in the Prow Pass tuff geologic unit. These tests will provide hydraulic parameters and sorption information from depth intervals that have not yet been intensively tested. Sorption measurements will be compared to laboratory sorption data to evaluate how well laboratory measurements represent field conditions at Yucca Mountain. This work will support Milestones M2HD and M2HF (Section 7).

3.1.3.2 Saturated Zone Hydrochemistry Data

Hydrochemistry data will be obtained using water samples collected from wells to the south of Yucca Mountain in Amargosa Valley, in cooperation with

Table 3-3. Application of Planned Technical Work for Saturated Zone Processes at the Site to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Post-closure Safety Case[1]	Elements of the Postclosure Safety Case[1]						Pre-closure Safety Case[3]	PSS Activities
		1. Expected Postclosure Performance		2. Design Margin/ Defense in Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Performance Confirmation		
		Principal Factors Supported by Technical Work	Priority to SR/LA[2]						
C-Wells Hydraulic Testing (3.1.3.1)	√	Formation and transport of radionuclide-bearing colloids Flow and transport in the saturated zone	2 1						2029 2245 8609
Saturated Zone Hydrochemistry Data (3.1.3.2)	√	Neptunium solubility Formation and transport of radionuclide-bearing colloids Flow and transport in the saturated zone	1 2 1						2029 2245 8609
Early Warning Drilling Program (Nye County) (3.1.3.3)	√	Flow and transport in the saturated zone	1				√		2029 2245 8609
Site-Scale Saturated Zone Flow and Transport Model (3.1.3.4)	√	Flow and transport in the saturated zone	1			√			2031 8611
Regional-Scale Saturated Zone Flow Model (3.1.3.5)	√	Flow and transport in the saturated zone	1						2031 8611

[1] Postclosure safety case described in Section 2.2.

[2] Postclosure work prioritization for site recommendation and LA (SR/LA) described in Section 2.2.4.

[3] Preclosure safety case described in Section 2.3.

[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

Nye County and the USGS, Nevada District; and wells on the Nevada Test Site, in cooperation with DOE Nevada. Oxidation/reduction potential, alkalinity, and pH will be measured in situ, and oxidation/reduction couples will be evaluated using laboratory chemical analysis for major, minor, and trace element abundance. This will work support Milestones M2HD and M2HF (Section 7).

3.1.3.3 Early Warning Drilling Program (Nye County)

Nye County, Nevada, in cooperation with DOE, will implement a program to install a series of wells in Amargosa Valley and the southern portion of the Nevada Test Site. The purpose of these wells is to monitor the saturated zone along possible transport pathways from Yucca Mountain for radionuclides that would be present in the inventory of the potential repository.

The first phase of the planned work will include constructing several new shallow and deep wells to measure hydraulic parameters for the alluvial and tuff aquifers. These wells would also allow any potentially contaminated water that would bypass the deep monitoring system to be detected.

Additional, deep monitoring wells are also planned. These wells will, over several years extending beyond submission of the LA, provide a comprehensive database of aquifer properties for the carbonate aquifer downgradient from Yucca Mountain. They will provide information to define the interconnection of the deeper carbonate aquifer with the shallower volcanic and alluvial aquifers, which will be used to characterize transport processes such as dilution. The deep monitoring wells will also serve to detect any contamination that might travel from a Yucca Mountain repository into the volcanic or carbonate aquifers.

As part of the cooperative agreement with Nye County, DOE will independently collect data and samples from these wells, consistent with the DOE quality assurance program. Early data collection for use in the LA will emphasize hydrostratigraphy, hydrochemistry, geophysical borehole logging, side-wall coring where necessary to support logging, aquifer testing, and measurements of oxidation/reduction potential and pH.

The planned work will support Milestones M2HD and M2HF (Section 7). Two types of saturated zone models will be developed: a site-scale flow and transport model, and a regional-scale flow model. These are discussed in the following sections.

3.1.3.4 Site-Scale Saturated Zone Flow and Transport Model

The site-scale model representing movement of water and transport of radionuclides in the saturated zone will be updated to include the following input:

- Revised regional hydrostratigraphic data from planned geologic mapping south of Yucca Mountain
- Hydraulic and transport testing results from previous and planned testing of the Bullfrog/Upper Tram and Prow Pass units at the C-wells borehole complex
- Results from ongoing hydraulic tests in existing borehole WT-24
- Regional hydrochemistry and isotopic results, including apparent groundwater age, oxidation/reduction potential, pH, and chemical analyses

Results from the model will be abstracted for use in the TSPA for the LA and for evaluation of model sensitivity and uncertainty. Model sensitivity analyses will include evaluating transport of radionuclides following a postulated early failure of one, or a small number of, separated waste packages. The results of mountain-scale flow and transport modeling will support Milestones M2MD and M2MF (Section 7).

Information from natural and man-made analogs will be used to build confidence in models of water movement through the saturated zone at site scale.

3.1.3.5 Regional-Scale Saturated Zone Flow Model

The regional model of water movement through the saturated zone will also be updated for the LA. This update will incorporate the following input:

- Revised regional hydrostratigraphic data from planned geologic mapping south of Yucca Mountain
- Hydrostratigraphy results from the Nye County drilling program
- Hydrostratigraphy results from boreholes at Yucca Mountain and associated with the Nevada Test Site Environmental Restoration Program
- Stratigraphic information obtained from existing geophysical survey data

Results from the model will be abstracted for use in the TSPA for the LA and for evaluation of model sensitivity and uncertainty. The results of this flow and transport modeling will support Milestones M2HD and M2F (Section 7).

The following activities take place after the cutoff dates for the above models but will be used as support for the LA or for confirmation of values used in the TSPA for the LA and the LA.

3.1.3.6 Downgradient Alluvial Hydraulic and Tracer Testing

An Amargosa alluvium testing complex will be established in cooperation with Nye County. The test will be designed to evaluate hydraulic parameters, effective flow porosity, longitudinal dispersivity, colloid transport parameters and sorption. Comparisons will be made to laboratory data to confirm the applicability of such data to those obtained in the field. This work will feed the saturated zone flow and transport model and the TSPA.

3.1.3.7 Enhance K_ds in Alluvium and Provide Support to the License Application

Transport in alluvium and K_d enhancement will be done on samples recovered from the new Nye County boreholes. The results will aid in the performance assessment abstraction and sensitivity analysis of the saturated zone flow and transport model after the models are delivered to performance assessment, and during the development of the TSPA for the LA. Transport issues resulting from the TSPA for the LA will be addressed as necessary and included in the LA. This activity will document the saturated zone flow and transport in alluvium for the LA.

3.1.3.8 Downgradient Volcanic Hydraulic and Tracer Testing

Hydraulic and tracer tests will occur at a volcanic downgradient site after flow paths from 5 to 20 km (3 to 12 miles) have been evaluated. These tests will be designed to evaluate aquifer parameters over a wide area along a flow path and obtain effective hydrologic properties averaged over a large volume of rock. Early data may be used in the LA to support assumptions made in the TSPA for the LA.

3.1.3.9 Demonstration of Total Concentration Reduction

Hydraulic and tracer tests at a downgradient site will continue to determine experimentally whether the total concentration of tracers is reduced. Tests will allow a direct measurement of the reduction in concentration of tracers as they travel from a point of release at the water table to a pumping well. This information will be used by performance assessment in saturated zone models after the LA is submitted to confirm performance and the appropriateness of the values used in the TSPA for the LA.

3.1.4 Thermal Testing

Work described by this group of work plans will use field thermal tests to address how heat affects the natural system. Field thermal tests are designed to improve confidence in the modeling of thermal, hydrologic, mechanical, and chemical coupled processes that will operate near the waste-emplacement drifts. This includes testing of the thermal and mechanical response of welded tuff, and the effects on movement of water and the dry-out and rewetting of the host rock.

Planned field thermal testing will directly address principal factors for the effects of heat and excavation on flow (drift-scale), and chemistry of water on waste packages, which are assigned relatively high priority for the LA, in Section 2.2 (Table 2-2). Field thermal tests will continue to contribute important supporting data for validating the coupled process models used for repository performance assessment and design.

Safety Case. This work will support the first element of the postclosure safety case, assessment of expected postclosure performance, by reducing uncertainty on how heat affects near-field processes. Models of these processes will provide input to assessments of long-term performance. Later stages of the planned field tests will also address the fifth element of the postclosure safety case, performance confirmation, because data will continue to be collected and analyzed after the cut-off date for inclusion in the LA.

NRC Key Technical Issues. The planned work will address various elements from five of the NRC key technical issues:

- Evolution of the Near-Field Environment (Section 4.3.3.3), particularly the subissues concerning effects of coupled processes on the rate of seepage into the repository, and on waste package lifetime

- Container Life and Source Term (Section 4.3.3.4), particularly the subissues related to the effects of corrosion on waste package lifetime, for humid air and aqueous corrosion, and also the effects of materials stability and mechanical failure on waste package lifetime, especially at elevated temperatures

- Thermal Effects on Flow (Section 4.3.3.5), particularly the subissues related to sufficiency of the thermal testing program and the thermal-hydrologic modeling approach, and also the aspects related to the adequacy of testing and incorporation of results into calibrated models

- Repository Design and Thermal-Mechanical Effects (Section 4.3.3.6), particularly the subissue related to consideration of thermal-mechanical effects on underground facilities design, and changes in hydrologic properties of fractures caused by thermomechanical perturbation of the rock

- TSPA and Integration (Section 4.3.3.7), especially the subissues related to model abstraction, data adequacy, and integration

Project Summary Schedule Activities. The work categories in the following sections will be completed to support the site recommendation and the LA. Each work category corresponds with one or more project summary schedule activities in the project budget and schedule, and many project summary schedule activities cover more than one work category. The project summary schedule activities which include work for thermal testing are 2033, Near-Field Environment Results to Support TSPA-SR; 2050, Testing for Enhanced Characterization of the Repository Block; 2253, Near-Field Environment Data and Analysis Update for SR/LA; 2270, Single Heater Test Cool Down; and 6107, Drift Scale Heater Test-Heat Up Phase.

These work categories are described in more detail in the following sections. The applicability of each work category to the postclosure and preclosure safety cases is summarized in Table 3-4. The "Priority to SA/LA" assessed in Section 2.2 is also shown on the table as a number from "0" to "3" (lowest to highest priority). The table lists the project summary schedule activities associated with each work category, for which cost and schedule information is given in Sections 6 and 7, respectively.

Table 3-4. Application of Planned Technical Work for Thermal Testing at the Site to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Postclosure Safety Case[1]	Elements of the Postclosure Safety Case[1]: 1. Expected Postclosure Performance — Principal Factors Supported by Technical Work	Priority to SR/LA[2]	2. Design Margin/ Defense in Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Performance Confirmation	Preclosure Safety Case[3]	PSS Activities[4]
Large Block Test (3.1.4.1)	√	Effects of heat and excavation on flow (drift scale)	2						2033 2253
		Humidity and temperature at waste package	0						
		Chemistry of water on waste package	2						
Single Heater Test (3.1.4.2)	√	Effects of heat and excavation on flow (drift scale)	2					√	2270
		Humidity and temperature at waste package	0						
		Chemistry of water on waste package	2						
Drift-Scale Heater Test (3.1.4.3)	√	Effects of heat and excavation on flow (drift scale)	2				√	√	6107
		Dripping onto waste package	2						
		Humidity and temperature at waste package	0						
		Chemistry of water on waste package	2						
Cross Drift Thermal Test (3.1.4.4)	√	Effects of heat and excavation on flow (drift scale)	2				√	√	2050
		Humidity and temperature at waste package	0						
		Chemistry of water on waste package	2						

[1] Postclosure safety case described in Section 2.2.

[2] Postclosure work prioritization for site recommendation and LA (SR/LA) described in Section 2.2.4.

[3] Preclosure safety case described in Section 2.3.

[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

3.1.4.1 Large Block Test

The large block test was heated for more than a year and will cool through September 1998. The post-cooling characterization will involve coring, overcoring, and laboratory analyses of mineralogic/petrologic and hydrologic properties. A detailed analysis of the data and reporting will be complete in August 1999 to support the near-field models milestone for the site recommendation and the LA (Milestone M2JC) and the site recommendation and LA design (Milestones M2MQ and M2MT).

3.1.4.2 Single Heater Test

During the single heater test, the rock mass was heated for 9 months and cooled for an equivalent period. Post-cooling characterization and detailed data analysis will be complete by September 1998. Reporting will be complete in March 1999 (Milestone M2FP) to support the near-field models Milestones M2JC, M2MQ, and M2MT (Verification of Requirements for SR/LA Design).

Planned work for Activity 6107, Drift Scale Heater Test-Heat Up Phase, will complete the drift-scale test, while Activity 6050, Testing for Enhanced Characterization of the Repository Block, will implement the cross drift test. Each is briefly described in the following sections.

3.1.4.3 Drift-Scale Heater Test

The drift-scale test heating phase started on December 3, 1997, and will continue for a total of 4 years. Automatic data collection and analysis of temperature, humidity, and mechanical deformation will continue throughout this period. Borehole testing, sampling of gas and water from the rock, and tomographic imaging will be conducted periodically. Reports will be prepared at regular intervals to publish up-to-date test results. A detailed report will be developed in September 1999 (Milestone M2JE), to support the near-field models milestone (Milestone M2JC) and the site recommendation and LA design (Milestones M2MQ and M2MT). Collection and analysis of data will continue during the heat-up phase until 2002 and will be used to evaluate the models and parameter values used in the TSPA for the LA and the LA.

3.1.4.4 Cross Drift Thermal Test

The cross drift thermal test will be designed and conducted in the cross drift constructed in 1998. It will address specific issues concerning thermal-hydrologic and thermal, hydrologic, and mechanical coupled processes in the proposed repository horizon. Information from the cross drift thermal test will provide important confirmatory data concerning thermal-hydrologic properties, water chemistry, thermomechanical deformation, and permeability changes for the site recommendation and the LA.

3.1.5 Near-Field Environment and Coupled Processes

These work plans will continue studies to address the effects of excavation and heat on the near-field environment and the altered zone. The principal goal is to improve confidence in predictive models that will be used to evaluate the impacts of thermally and chemically driven coupled processes on natural and engineered barrier performance. Model development will rely on results from thermal tests in both the laboratory and the field (see Section 3.1.4).

Recent studies have shown that thermohydrologic models can predict temperature and hydrologic conditions in the host rock, accounting for fracture effects on a spatially averaged basis. For example, reasonable agreement has been obtained between model results and observations from the large block test and the single heater test. In addition, these field tests have provided observational data on coupled processes, including mechanical response and the evolution of water chemistry. It is expected that results from the drift-scale test will also be important in substantiating predictive model capabilities.

Laboratory work has provided chemical data and rock properties for use in modeling and has provided insights into the importance of coupled processes. Recent studies have shown that coupled thermal, hydrologic, and chemical effects may be important on time scales that are not readily accessible by field testing. Predictive simulation will be heavily relied on in investigating long-term thermal, hydrologic, chemical, and mechanical coupled processes, to understand the impacts on barrier performance. Controlled laboratory investigation of coupled processes, as well as natural analogs, will be emphasized to help validate models for this purpose.

The planned work will address principal factors for effects of heat and excavation on flow (drift scale), dripping onto waste packages, chemistry of water on waste package, and formation and transport of radionuclide-bearing colloids. All of these factors are assigned relatively high priority for the LA in Section 2.2 (Table 2-2). The principal factors for temperature and relative humidity on waste packages, and transport through and out of the engineered barrier system, will also be addressed (Table 3-5).

Safety Case. Studies on the near-field environment and coupled processes will support the first element of the postclosure safety case, assessment of expected postclosure performance. Updated,

Table 3-5. Application of Planned Technical Work on the Near-Field Environment and Coupled Processes to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Post-closure Safety Case[1]	Elements of the Postclosure Safety Case[1]						Pre-closure Safety Case[3]	PSS Activities[4]
		1. Expected Postclosure Performance		2. Design Margin/ Defense in Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Performance Confirmation		
		Principal Factors Supported by Technical Work	Priority to SR/LA[2]						
Simulate the Performance of Design Options for the Engineered Barrier System (3.1.5.1)	√	Effects of heat and excavation on flow (drift scale) Dripping onto waste packages Humidity and temperature at waste package Chemistry of water on waste package Formation and transport of radionuclide-bearing colloids Transport through and out of the engineered barrier system	2 2 0 2 2 1	√		√			2035 2253 8615
Comparative Testing of Candidate Materials (3.1.5.2)	√	Chemistry of water on waste package Formation and transport of radionuclide-bearing colloids	2 2			√	√		2035 2253 8613 8615
Mechanical Stability of Ground Support Materials (3.1.5.3)	√	Chemistry of water on waste package Formation and transport of radionuclide-bearing colloids	2 2						2035 2253 8613 8615
Thermal-Hydrological-Mechanical Coupled Effects (3.1.5.4)	√	Effects of heat and excavation on flow (drift scale) Seepage into drifts Chemistry of water on waste package	2 3 2			√			2033 2035
Thermal-Hydrological-Chemical Coupled Effects (3.1.5.5)	√	Effects of heat and excavation on flow (drift scale) Seepage into drifts Dripping onto waste packages Humidity and temperature at waste package Chemistry of water on waste package	2 3 2 0 2			√			2033 2035 2253 8613 8615

[1] Postclosure safety case described in Section 2.2.

[2] Postclosure work prioritization for site recommendation and LA described in Section 2.2.4.

[3] Preclosure safety case described in Section 2.3.

[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

validated, and qualified models of near-field and altered-zone processes will be used to assess long-term performance. The same models will also be used to evaluate design features and options such as backfill, drip shields, and a Richards' barrier.

NRC Key Technical Issues. Results from this work will address the NRC key technical issue for Evolution of the Near-Field Environment (Section 4.3.3.3), including subissues concerned with the effects of coupled processes on seepage, waste package lifetime, the rate of radionuclide release from breached waste packages, and radionuclide transport through engineered and natural barriers. The planned work will address changes in hydrologic properties of rock fractures and matrix caused by thermal, hydrologic, and chemical coupled processes, and the long-term effects of introduced materials such as concrete. The results will address aspects of five other key technical issues:

- Container Life and Source Term (Section 4.3.3.4), will be addressed through development of model capabilities to predict conditions that produce humid air corrosion and aqueous corrosion.

- Thermal Effects on Flow (Section 4.3.3.5) will be addressed, particularly the subissues for assessing the potential for thermal reflux to occur and predicting the bounds on thermal effects on flow.

- Repository Design and Thermal-Mechanical Effects (Section 4.3.3.6) will be addressed, particularly the subissue related to thermal-mechanical effects on underground facility design and performance. The planned work will address changes in hydrologic properties of rock fractures caused by thermal-mechanical perturbation of the rock mass that might adversely affect the near-field environment.

- TSPA and Integration (Section 4.3.3.7) will be addressed, particularly the subissue related to model abstraction, data adequacy, and integration.

- Radionuclide Transport (Section 4.3.3.10) will be addressed, especially the subissues concerning geochemical and hydrological controls on radionuclide transport, and the sensitivity of overall system performance to ranges of parameters that control transport.

Project Summary Schedule Activities. The work categories in the following sections will be completed to support the site recommendation and the LA. Each work category corresponds with one or more project summary schedule activities in the project budget and schedule, and many project summary schedule activities cover more than one work category. The project summary schedule activities that include work for near-field environment and coupled processes are 2033, Near-Field Environment Results to Support TSPA-SR; 2035, Near-Field Environment Results to Evaluate Waste Package Lifetime and Engineered Barrier System Transport for SR/LA; 6105, Science Support to License Application; 8613, Confirm Near-Field Environment Data Results for Construction Authorization; and 8615, Confirm Near-Field Environment Modeling Conclusions for Construction Authorization.

These work categories are described in more detail in the following sections. The applicability of each work category to the postclosure and preclosure safety cases is summarized in Table 3-5. The "Priority to SR/LA" assessed in Section 2.2 is also shown on the table as a number from "0" to "3" (lowest to highest priority). The table lists the project summary schedule activities associated with each work category, for which cost and schedule information is given in Sections 6 and 7. Each of the work categories described in the following sections will support Milestones M2JC, M2MQ, and M2MV (Section 7). Work will continue past the dates of the milestones to test the sensitivity of model results to uncertainties in input parameters. This work will provide added confidence in the information contained in the LA.

3.1.5.1 Simulate the Performance of Design Options for the Engineered Barrier System

Work in this category will support evaluating design options and alternatives. Design options for the engineered barrier system and alternatives, such as backfill, drip shields, and Richards' barriers, will be simulated to evaluate how sensitive the performance of the engineered barrier system is to the uncertainty in hydrologic properties of the host rock and the materials used for engineered barriers. The simulations will also investigate the likely effects from future changes in hydrologic boundary conditions and degradation of materials with time.

3.1.5.2 Comparative Testing of Candidate Materials

This task will provide data for comparative analysis of alternative materials in the LA (10 CFR 60.21). For example, ground support materials identified for the VA reference design include steel and concrete. Testing to directly support the LA will commence in 1999 after the LA reference design decision identifies specific candidate materials. Testing will evaluate physical and chemical changes in response to heating, representative flow and transport properties, and the effects of nutrients conducive to microbial growth. The planned work will include both short-term and long-term multi-year tests.

3.1.5.3 Mechanical Stability of Ground Support Materials

Work on this task will evaluate the long-term mechanical stability of the host rock, and of candidate ground support materials, based on laboratory test results and natural or man-made analogs. The long-term effects on the ground control system from thermal-mechanical loading and unloading of the host rock will also be investigated. This will also predict the nature and timing of ground support collapse, and the effects on waste packages and other barriers.

3.1.5.4 Thermal, Hydrological, and Mechanical Coupled Effects

This work will address the effects of thermal, hydrological, and mechanical coupled processes. This task will forecast the mechanical stability of the host rock in response to heating, emphasizing the inelastic behavior of fractures, and compare the predictions with the observed results from microseismic monitoring of the drift scale test. The hydrologic response of fractures at representative locations within the host rock will also be evaluated as function of time. This work will result in coupled models used to evaluate the sensitivity of thermal and post-thermal flow regimes to the mechanical effects of heating. In addition, predictive modeling will be carried out to support the abstraction of process models of host rock response for use in performance assessment.

3.1.5.5 Thermal, Hydrological, and Chemical Coupled Effects

Work in this category will address the effects of thermal, hydrological, and chemical coupled processes. Numerical models will be developed to represent such fully coupled processes to evaluate their effects on waste package lifetime, rates of radionuclide release, and radionuclide transport. The predictive capabilities of the models will be tested using laboratory experiments and simulations of natural analog observations. Refined model abstractions will be provided to bound the effects of thermal, hydrological, and chemical coupled processes on natural and engineered barrier system performance. The planned work will result in fully coupled thermal, hydrological, and chemical models that will contribute to resolution of the NRC key technical issue on Evolution of the Near-Field Environment (Section 4.3.3.3).

3.1.5.6 Evolution of the Near-Field Geochemical Environment

This task will develop mechanistic chemical model predictions of the composition of the water contacting the waste packages and other engineered barriers, and the effects of introduced materials in the near-field geochemical environment. The behavior of mobile reactants in the near-field geochemical environment will be coupled with thermal-hydrologic behavior and mountain-scale convective processes. Bounding analyses will be developed for the accumulation of biomass in the near-field environment that can affect the corrosion of the waste packages and contribute to release and transport of radionuclides. This work will result in coupled models that incorporate the effects of liquid-phase and gas-phase transport of reactive components, oxygen, carbon dioxide, and water, that interact with introduced materials such as concrete. Models for long-term evolution of the composition of water in the near-field geochemical environment will thus be coupled with drift-scale and mountain-scale thermohydrologic and thermal, hydrologic, and chemical models.

3.1.5.7 Altered Zone Flow and Transport Model

The planned work will use laboratory experiments and modeling to determine the extent to which drainage of warm water from the repository horizon could alter the lowermost Topopah Spring tuffs. Chemical alteration of these glassy rock units could cause transport pathways to bypass much of the Calico Hills unit, and thus limit the potential for radionuclide retardation in an important natural barrier. Experiments and models will evaluate the extent to which fracture-matrix interaction is likely to be permanently altered by thermal, hydrologic, and chemical coupled processes. Additional modeling work will assess the potential for mineral precipitation on fracture surfaces. The results of this planned work will be bounding analyses of the effects of coupled processes on fracture-matrix interaction, fracture permeability, and radionuclide transport.

3.1.6 Performance Confirmation

The purpose of performance confirmation is to provide additional assurance that models and data used in assessing preclosure and postclosure performance assessment are reasonable. The need for performance confirmation is identified in 10 CFR 60 as a component of the LA, and performance confirmation is one element of the postclosure safety case described in Section 2.2.1. The planned work will include monitoring water levels and earthquake activity, assessing the performance of seals, and support to systems engineering for maintenance and oversight of the performance confirmation plan, participation in systems studies, and support for testing and evaluation activities. Specific tasks include the following:

- Stream flow data will continue to be monitored to document major and extreme runoff events.

- The Southern Great Basin Digital Seismic Network will be maintained and operated to document seismic events. A network of nine strong-motion instruments will also continue to operate to provide information on ground motion from large earthquakes. Results will be used to confirm that inputs used for seismic design are appropriate.

- In situ tests of prototype seal components will be performed in fiscal years 2000 and 2001. These tests will provide initial baseline data for performance of the seal designs used in the LA. Factors including emplacement techniques, initial strength, and permeability will be evaluated to assess baseline performance.

- Systems engineering will develop design criteria, resolve design assumptions, and identify site related impacts for program enhancements such as repository expansion. These activities will support verifying the LA design satisfies all requirements. Work will also specifically support test and evaluation activities.

The project summary schedule activity that includes the currently planned performance confirmation work is 7027, Performance Confirmation Data Collection. A performance confirmation plan based on the LA reference design to be selected in 1999, will be prepared in conjunction with the LA, and may include additional technical work.

3.1.7 Management and Integration, Report Preparation and Review, and Field Support

Several management and support activities will be maintained to ensure successful completion of the site characterization program, and support of the LA and performance confirmation. The project summary schedule activities which include this work are 8621, Test Coordination and Support for Site Activities; 9090, Site Investigations Base Support; and 6105, Science Support to License Application. These activities will cover three broad areas, as described in the following sections.

3.1.7.1 Test Coordination and Support for Site Activities

This planned effort will consist mainly of coordinating scientific and engineering testing at the Exploratory Studies Facility and the Busted Butte facility. These tests have been described in the preceding sections and will be conducted with specific objectives related to the preclosure and postclosure safety cases. Test coordination will include planning and oversight for specific testing projects, staffing, and procuring equipment.

3.1.7.2 Site Investigations Base Support

This support function will consist of management and integration of projects, schedules, and results with various organizations. Work in this area will include the following:

- Integrate technical results from site investigations, with design activities and products.
- Collaborate with appropriate organizations, internal and external to the project, to share ideas and review progress.
- Develop long-range plans and cost estimates.
- Monitor the progress of ongoing activities.
- Prepare routine reports, including semiannual site characterization progress reports required by the Nuclear Waste Policy Act (as amended).
- Qualify data, software, and reports that are required for performance assessment or safety-related design functions.

3.1.7.3 Science Support to the License Application

This work will consist of preparing the site characteristics section of the LA and supporting the preparation of other sections. Supporting the preparation of the LA also includes evaluations of natural analogs for the processes operating at Yucca Mountain to provide greater confidence in site modeling results, and evaluation of new data that is developed outside the project for their implications to previous conclusions and assessments. It will also include scientific support for preparation of the EIS.

3.2 DESIGN

Design activities have been planned to obtain the information identified in Volume 2 and in Section 2 of this volume. This section describes the planned design activities, and relates them to the preclosure and postclosure safety cases, and the NRC key technical issues.

Common to each of the following sections is a set of activities necessary to establish the bases for the designs to be pursued and to verify that the resultant designs have met these bases. A concerted effort will continue in 1999 to define the systems, structures, and components that comprise the Monitored Geologic Repository; the functions these systems must perform to accomplish the overall mission; and the criteria that define how well these systems must perform their functions. Multi-organization teams involving engineers, scientists, and modeling personnel will perform studies, analyses, and trade-off evaluations to determine the appropriate requirements applicable to the systems, structures, and components. These requirements will then be documented in a series of system description documents for use in subsequent design analyses. Compliance with regulatory requirements and applicable codes and standards is mandated. Sections 3.2.1, 3.2.2, and 3.2.3 illustrate how the designs resulting from the activities meet the requirements.

The most comprehensive of these evaluations, selecting the LA referenced design, is currently underway and will be completed in 1999. This selection process, involves quantitatively evaluating design alternatives, features, and options, as described in Volume 2, Section 8. The results of this evaluation will not only identify the initial design to be the basis for the site recommendation and the LA, but will also provide a basis for quantitative allocation of postclosure performance requirements and functions to the systems, structures, and components which comprise the Monitored Geologic Repository. Completion of this effort will culminate in a series of milestones (M2HW, M2HY, and M2MU) in which comprehensive verification of requirements is performed to ensure the requirements are reasonable and complete, that they have a valid basis and justification, and that they trace back to the top level requirements levied on the Monitored Geologic Repository.

The results of the design activities described in Sections 3.2.1, 3.2.2, and 3.2.3 will be captured, in a summary fashion, by the system description documents. The design features and alternatives evaluations will be used to support an initial selection of reference design for the site recommendation and the LA (Milestone M2MQ), followed by more detailed design analyses to support the safety evaluations required by the LA process (Milestones M2MX and M2MV).

As the design solutions are developed, compliance with the previously established requirements will be verified. Comprehensive verification of the design to the requirements is the focus of Milestones M2MU and M2MW. Performance assessment efforts defined in Section 3.3 will support design verification for the postclosure performance requirements. Finally, verification of the as-built system will be the subject of test and evaluation programs, defined as a series of inspections, tests, analysis, and acceptance criteria which will define the testing, examinations, demonstrations, and analyses necessary to illustrate compliance of the as-built system with the design and requirements.

3.2.1 Subsurface Design

The subsurface design work will support preclosure and postclosure performance. The work plans in this section reflect the DOE plans and are organized into 13 functional categories and categorized using a process called "binning." Binning categorizes systems, structures, and components relative to radiological safety and waste isolation, and precedent in licensing. Bin 3 has radiological safety implications with no licensing precedent. Bin 2 has radiological safety implications, but also has licensing precedents. Bin 1 has no relation to radiological safety. Binning is described in detail in Volume 2, Section 2.3.

Work scheduled for the different functional categories will support the postclosure and preclosure safety cases (as outlined in Section 2), and address the balance of plant design needed for the site recommendation and the LA. Systems, structures, and components that are individually non-safety related could impact the performance of other systems, structures, and components that deal with radiological safety during preclosure and postclosure phases. In addition, these designs must be fairly well understood so that information for estimating costs can be provided.

Models developed to support the safety cases will rely heavily on information from subsurface design. This information is particularly important when evaluating the performance of materials to be used in construction (e.g., when addressing the principal factor for the formation and transport of radionuclide-bearing colloids), the arrangement of waste packages (for evaluating thermal management effects on flow and lifetime of engineered barriers), the physical placement and excavation of underground openings (e.g., for modeling seepage into drifts), and the feasibility of drip shields and other engineered features or design alternatives.

Information on the structural characteristics of the rock and the design of the repository subsurface helps to identify design basis events, and leads to understanding what measures should be taken in the design to reduce the potential, where possible, for those events to occur. Activities taking place during the preclosure phase of the project (e.g., material degradation of permanent items and accidental spills) could have long-term impact on the postclosure performance.

This section discusses the 12 functional areas listed below. Each area is also described in Volume 2, Sections 4, 5, and 6:

- Subsurface layout
- Ground control

- Subsurface ventilation
- Waste emplacement
- Subsurface monitoring and control systems
- Waste package retrieval
- Performance confirmation
- Sealing and closure
- Subsurface utilities
- Subsurface integrated control systems
- Radiological safety
- Engineered barrier system performance modeling and testing

Table 3-6 shows the relationships between these areas, the preclosure and postclosure safety cases, and the references to cost and schedule information. Project summary schedule activity numbers are given for the categories of work described in the text.

All systems described in this section will use demonstrated technology where practical, meet NRC requirements and accepted design criteria, and as appropriate, withstand the effects of design basis events.

As a part of the design process, and in compliance with regulatory requirements, a comprehensive quantitative evaluation of additional design features and fundamental design alternatives is underway and will be completed in 1999. Section 8 of Volume 2 discusses the preliminary evaluation of alternatives. To the extent possible, the planned study of alternatives will use existing information to evaluate alternative designs, design features, and options. However, additional near-term efforts are anticipated to develop new information to be used for the comparisons. These efforts will focus on five alternative categories and attendant features, as follows:

Category 1. Containment in the Engineered Barrier System

- Barriers
  - Drip shield
  - Backfill
  - Richards' barrier
  - Diffusive barrier or getter under waste package
- Emplacement mode

Category 2. Other Engineered Enhancements

- Drift linings and ground control systems
- Rock treatment

Category 3. Integrated Effects of Thermal Loading

- Thermal loading and thermal management
- Ventilation
- Drift diameter

Category 4. Waste Package Production and Emplacement Operations

- Shielding and accessibility

Category 5. Deferred Closure

- Underground features
- Timing

Initial evaluations and data collection will focus on obtaining information pertinent to the evaluation criteria and decision methods to be used in the alternative trade-off evaluations. Information generated in the subsurface design area will generally involve feasibility of the concepts, cost impacts, flexibility, schedule, and preclosure radiological safety. Performance implications associated with the features and alternatives will generally be determined by performance assessment calculations.

Table 3-6. Application of Planned Subsurface Design Work to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Post-closure Safety Case[1]	Elements of the Postclosure Safety Case[1]: 1. Expected Postclosure Performance: Principal Factors Supported by Technical Work	1. Expected Postclosure Performance: Priority to SR/LA[2]	2. Design Margin/ Defense in Depth	3. Dis-ruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Per-formance Confirmation	Pre-closure Safety Case[3]	PSS Activities[4]
Subsurface Facility (3.2.1.1)	√	Effects of heat and excavation on flow (mountain scale) Humidity and temperature at the waste package	1 0	√			√		2035 2253 8615 9090
Ground Control (3.2.1.2)	√	Chemistry of water on waste package Formation and transport of radionuclide-bearing colloids	2 2	√				√	2035 2253 8613 8615
Subsurface Venti-lation (3.2.1.3)	√	Effects of heat and excavation on flow (drift scale) Dripping on waste packages Humidity and temperature at the waste package	2 2 0	√				√	2035 2253 8613 8615
Waste Emplacement (3.2.1.4)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion resistant waste package barrier	1 3	√				√	2033 2035
Subsurface Safety and Monitoring (3.2.1.5)		N/A		√				√	2033 2035
Waste Retrieval (3.2.1.6)		N/A		√				√	2282 2383
Performance Confirmation (3.2.1.7)		N/A					√		2383 2387
Sealing and Closure (3.2.1.8)	√	Formation and transport of radionuclide-bearing colloids Transport through and out of engineered barrier system	2 1	√			√		2383 2387
Subsurface Utilities (3.2.1.9)		N/A		√			√	√	2383 2387
Subsurface Integrated Control Systems (3.2.1.10)		N/A		√				√	2382 2383 2387
Radiological Safety (3.2.1.11)		N/A		√				√	2382 2383 2387
Engineered Barrier System Perfor-mance Modeling and Testing (3.2.1.12)		Effects of heat and excavation on flow (drift scale) Dripping onto waste packages Humidity and temperature at waste package Chemistry of water on waste package Formation and transport of radionuclide-bearing colloids Transport through and out of engineered barrier system	2 2 0 2 2 1	√					2382 2383 2387

[1] Postclosure safety case described in Section 2.2.

[2] Postclosure work prioritization for site recommendation and LA (SR/LA) described in Section 2.2.4.

[3] Preclosure safety case described in Section 2.3.

[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

At the same time, subsurface design will move forward on developing features of the VA reference design, which are not expected to change in the selection process for the LA.

All of the work described in the following sections coincides with the schedule of activities and milestones presented in Section 7. Activities for completing the work will be planned in detail during the fiscal year planning effort to ensure that all necessary work to support the planning milestones is completed according to schedule.

3.2.1.1 Subsurface Facility

Work in this area will support Milestones M2MQ, M2MU, M2MX, and M2MV presented in Section 7. This work scope will include design analyses, detailed drawings of the general arrangement and layout of the underground facility, and equipment specifications and descriptions. The items that will be completed as a result of this work are as follows:

- Assessments of how the subsurface design may be affected by specified physical characteristics and design conditions (Bin 3)
- Schedules for constructing the subsurface repository and emplacing the waste packages (Bin 2)
- Descriptions of the arrangement of the waste packages and the strategy for their emplacement (Bin 3)
- General arrangements for surface facilities that support the subsurface operations, such as shaft head frames and hoisting systems, and special handling and storage facilities (Bins 1 and 2)

In addition, specific parameters and anticipated prevailing conditions will be evaluated for how they impact subsurface layout. For example, the method for emplacing the waste packages will be evaluated, as will the method for retrieving them, if necessary. Other conditions that impact subsurface design are the characteristics of the waste being shipped to the site, the design of the containers that will hold the waste, and the amount of heat generated by the waste packages after they are inside the emplacement drifts (thermal management).

Spacing of the waste packages within the emplacement drifts and the placement of the drifts themselves are tied closely to the thermal management strategy. Spacing for the emplacement drifts and for each waste package type and size will be specified and described. Making these decisions involves evaluating results from thermal-mechanical models, field thermal tests and coupled-process models, as well as assessing the effects of various measures to enhance the long-term performance of the repository. Results from modeling a range of potential drift sizes, spacings, thermal loads, and waste package arrangements will be evaluated to support selecting the LA design.

Other areas that will also be considered include the following:

- Components that enhance the reference and alternative engineered barrier systems
- Requirements for shielding and drainage
- Access to the emplacement drifts
- Temperature limits
- Performance confirmation plans
- Design basis events

The work scope for subsurface layout will support the postclosure safety case by designing measures to control the effects of water, accommodating the components of the engineered barrier system, and planning for specialized performance testing. Specifying the spacing between waste packages supports selecting the thermal-management strategy. This strategy will be selected to minimize or delay water seeping into emplacement drifts during the thermal and post-thermal (ambient temperature) periods and delay the return of humidity to ambient conditions. Keeping the humidity low can contribute to longer lifetimes for the waste packages. The subsurface layout design will also support enhancements to the engineered barrier system, such as drip shields or backfill in the drifts. The subsurface layout will reflect any special testing needs of the performance-confirmation program that is part of the postclosure safety case.

The work planned for subsurface design will contribute to the preclosure safety case. Underground ventilation systems, filters, shielding systems, and any other systems determined by analysis to be important to radiological safety will be designed.

One of the NRC key technical issues is Repository Design and Thermal-Mechanical Effects. Two subissues for this topic are design of the proposed repository to meet preclosure and postclosure performance objectives and evaluation of thermal effects on the design of the underground facility. The work planned for this phase of subsurface design addresses both of these subissues by implementing the selected thermal management strategy and evaluating thermal-mechanical effects.

3.2.1.2 Ground Control

Work in this area will support Milestones M2MQ, M2MU, M2MX, and M2MV presented in Section 7. The ground control work scope will include designing systems that support and restrain the structural rock mass, called ground control, for subsurface facilities and openings. Rockbolts and wiremesh, or concrete lining, are examples of ground control systems. This activity addresses a Bin 3 system and its components.

Both temporary and permanent ground control systems will be evaluated for use in emplacement-side and development-side locations and accesses. These evaluations consider the following:

- Environmental, seismic, and thermal conditions
- Materials used
- Maintenance required
- Longevity of the systems and components, including deferred closure cases
- Construction interfaces

The work scope is outlined as follows:

- Refining the designs of support and stabilization systems for the layout, using the most recent information from in situ and laboratory testing
- Analyzing the performance of materials under the range of anticipated conditions during the preclosure period, that is, while the repository is operating
- Evaluating the longevity of materials, such as concrete and steel products, for use in preclosure and postclosure phases for all of the repository subsurface openings, as well as supporting evaluations of the material compatibility and performance interactions with other elements in the system
- Evaluating additional concepts that support studies of the design alternatives, such as unlined drifts, different drift sizes, and the treatment of near-field rock

This aspect of the design process will develop information that supports the postclosure safety case relative to characteristics of materials to be permanently emplaced underground. The characteristics of these materials and their structural performance have been identified as contributing to postclosure performance in the areas of ground support degradation, effects of rockfall, and the ability to emplace optional barriers. As mentioned in Section 2, degradation of ground support that leads to rockfall, may contribute to changes in the thermal-mechanical and seepage characteristics of the host rock. This information will be input to models that will estimate expected performance, aid in selecting relevant natural analogs, and estimate the consequences of disruptive processes and events. This information will be used in calculating margins of safety and evaluating defense in depth. This design provides input to systematically identifying and evaluating design basis events to classify systems, structures, and components based on safety.

As with subsurface layout, the ground control work will help to resolve the NRC key technical issue regarding Repository Design and Thermal-Mechanical Effects by providing information on the stability of the drifts. This information will, in turn, feed directly into the evaluation of thermal management, the retrievability of the waste packages, and potentially disruptive faults. The scope also addresses two additional key technical issues, Structural Deformation and Seismicity and Evolution of the Near-Field Environment, by considering seismic motion in the design and the impact of mechanical and chemical changes in materials used for ground support in the emplacement drifts. The planned work on the stability of the drifts and the retrievability of the waste packages specifically addresses aspects of the Repository Design and Thermal-Mechanical Effects key technical issue related to consideration of thermal-mechanical effects on underground facility design.

3.2.1.3 Subsurface Ventilation

The work in this area supports Milestones M2MQ, M2MU, M2MX, and M2MV as presented in Section 7. The design for the repository subsurface ventilation system will cover development activities and emplacement operations from the start of construction through facility closing. The existing design documents are used as a starting basis for this task. Assigned a Bin 3 designation, the subsurface ventilation system must be closely integrated with development and operational functions.

Ventilation design for development activities will cover temporary systems that must be installed to support preparing the repository to receive waste packages. The design will support excavating and equipping the main tunnels, exhaust main, shafts, initial emplacement drifts, and ancillary openings, as well as the development activities that are concurrent with emplacement operations. Construction activities include excavating and equipping the emplacement drifts and equipping the main drifts.

The ventilation system will provide a safe, workable environment in all areas that are accessible by personnel during waste emplacement. The emplacement side ventilation design will address systems for removing ambient moisture and radioactive gases and particulates potentially generated by the waste packages (caused by some off-normal event), and controlling changes in temperature. The ventilation design will include the operating and monitoring periods, as well as activities dealing with retrieving the waste packages, backfilling the drifts, and closing the repository. Both normal and off-normal situations will be considered for all of the repository's operating modes.

Designing the ventilation system will include the following:

- Assessing requirements, developing ventilation networks, and defining monitoring and control systems

- Identifying and describing major ventilation equipment including fans, control devices, duct work, stoppings, doors, and air cleaning systems

- Producing drawings, supported by analyses, for volumes and quality of air and performance of systems, structures, and components within the total system

Information on design basis events will support the postclosure safety case by supplying input to modeling that will estimate expected performance. The data will be used in calculating margins of safety and evaluating defense in depth. The data can provide meaningful input for reducing the concentration of radionuclides in the repository air stream caused by design basis events that have been identified to cause a potential release of radionuclides.

The connection of this work to preclosure safety is apparent in effectively controlling the flow of air underground. The quality and quantity of air in both the operations and development areas must be suitable for the requirements of each area and will work in continual balance to satisfy safety requirements relative to design basis events. This design function will provide input to systematically identifying and evaluating design basis events to classify systems, structures, and components based on safety.

The work scope for subsurface ventilation will provide information to resolve the key technical issue of Repository Design and Thermal-Mechanical Effects, specifically in the area of drift stability. This information will feed directly into the evaluation of thermal management strategies, retrievability of the waste packages, and long-term accessibility to the waste packages. The scope of work also addresses the key technical issue of the Evolution of the Near-Field Environment by considering the long-term use of ventilation for controlling temperature and humidity in the subsurface facility.

3.2.1.4 Waste Emplacement

Work in this area will support Milestones M2MQ, M2MU, M2MX, and M2MV presented in Section 7. The waste emplacement work scope will include all the work involved in designing the underground system for handling the waste packages, as well as the conceptual designs necessary to evaluate alternative techniques for emplacing the waste. The subsurface waste package handling system will transport waste packages from the Waste Handling Building on the surface to the underground emplacement area and will place them in the emplacement drifts. This dual-purpose system is classified as Bin 3.

Designing this system will require input from several technical disciplines, such as electrical, structural, remote handling, and nuclear safety engineering. Though these contributing elements are classified as Bin 2, they are essential to understanding completely, the function and safety of the total system.

The currently planned transport subsystem will include the rail system used to move the waste packages from the surface facilities to the underground emplacement area. Items to be designed or selected include the tracks and associated switches to drift turnouts, signals, and related mechanisms and control systems. Other mechanical items will include the rolling stock, such as a waste package transporter, locomotives, and rail cars. Interfacing systems will include communications, electric power supply, drift inverts, and other features and equipment directly associated with moving the waste packages.

The emplacement subsystem will focus on designing equipment, such as the gantry in the current design, used to maneuver the waste packages underground. Items to be designed or selected include the following:

- The gantry (current reference design) and associated mechanisms and control systems
- Other specialized items such as radiation shields and isolation doors at the entrances of drifts to separate the radiological environment from the transportation environment
- Interfacing systems including communications, electric power supply, and drift inverts

Items to be developed as a result of this work will include conceptual renderings of alternative systems, as well as drawings and preliminary specifications for the selected system. In addition, the level of detail planned for each selected subsystem, structure, and component will match the appropriate binning designation.

The work planned for subsurface waste package handling will support the preclosure safety case. It will emphasize the use of demonstrated technology, use accepted criteria, and provide input to the systematic identification and evaluation of design basis events to classify systems, structures, and components based on safety.

The designs that are developed for waste emplacement will deal directly with the handling and integrity of the waste packages, and are therefore safety related without having a precedent in a disposal environment.

The information developed for design in this area may show that residual effects from waste emplacement contribute to understanding postclosure performance by introducing materials that remain in the emplacement drift after the repository is closed. These materials could include signal cables, monitoring devices, and gantry rails on the inverts of the emplacement drifts. These materials and their characteristics will be identified and provided as input to performance assessment models. Work in this area will also support evaluating the key technical issue of Evolution of the Near-Field Environment because emplacement options will control thermal management, which in turn, will influence its geochemistry and hydrology.

The work scope addresses the key technical issue of Repository Design and Thermal-Mechanical Effects by providing information on emplacing and retrieving the waste packages. Because this application has no precedent in licensing, adequate design for this system is essential to fully understanding its operational functions and relative safety, particularly as they relate to the site recommendation and the LA.

3.2.1.5 Subsurface Safety and Monitoring

Work in this area supports Milestones M2MX, and M2MV presented in Section 7. This effort focuses on developing designs for the safety and monitoring systems used underground during both development and operational phases, including development, emplacement, retrieval, monitoring, backfill, and closure modes. Monitoring and control systems for the operations area, are classified as Bin 3; those in the development area, as Bin 2.

Safety and monitoring systems encompass designs used in the following:

- Transporting and emplacing waste packages
- Transporting other materials underground
- Maintaining personnel and radiological safety
- Monitoring excavation and ground control systems

Monitoring and control system design for operations will include waste handling, personnel and radiological safety, ground control, and criticality. Monitoring fire detection and ventilation systems is a part of this functional area.

This work scope will entail developing a plan, the processes, and an integrated architecture for the repository total control system. A cross-functional team will coordinate and integrate the level, type, and relationship among the various control systems for the repository. These systems control a variety of activities that are essential for the efficient and safe performance of the repository, including the following:

- Emplacement
- Rail transportation
- Ventilation
- Drift access
- Data, voice, and video communication
- Environmental and performance monitoring
- Power source and distribution
- Utilities
- Safety
- Security and plant protection
- Emergency response

Products of this effort will include drawings and preliminary specifications that are supported by analyses. The level of detail planned for each system, structure, and component coincides with the corresponding binning designation, which will be influenced by the selected approach for the LA design. The information developed in this work will be integrated with efforts in performance confirmation, repository operations, radiological safety, and industrial safety. Key interfaces exist with other specific design areas by coordinating designs for equipment and systems that will perform under expected conditions.

Input from this effort to show the use of demonstrated technology in designs will support the preclosure safety case. The designs will indicate that the design criteria accepted for this system either have been, or can be, satisfied using available technology. This design will provide input to systematically identify and evaluate design basis events to classify systems, structures, and components based on safety.

Normal operation standards will be demonstrated to show satisfactorily that this system can function consistently, be relied upon, and be maintained safely and efficiently.

Demonstrating the techniques envisioned for monitoring the performance of repository systems will support the postclosure safety case. These measures will provide key information on the performance of ground support systems and the preclosure environment. Both of these indicators are important in establishing and tracking postclosure key factors on ground support degradation, and effects of heat on near-field systems.

This work package will provide information to resolve the key technical issue of Repository Design and Thermal-Mechanical Effects, specifically in the areas of demonstrated technologies and safety during the construction, operation, and monitoring phases of the repository.

3.2.1.6 Waste Retrieval

Work in this area will support Milestones M2MQ, M2MX, and M2MV presented in Section 7. This work scope will focus on refining the designs and specifications for equipment and mechanical components used to retrieve the waste packages after they have been emplaced, considering normal and off-normal conditions. According to 10 CFR 60, waste packages must be retrievable for up to 50 years after the start of emplacement. The VA reference design is based on a 100-year retrievability period and longer periods, up to 300 years, are being considered. The reference design is based on the assumption that emplacement drift backfill will not be used. If emplacement drift backfill is ultimately used, it will only be placed after the decision has been made to close the facility. Retrieval is not planned after backfill placement.

Under normal conditions, the same major equipment will be used for emplacement and retrieval. Retrieving waste packages under off-normal conditions will require special equipment such as heavy-duty lifts, equipment carriers, waste package transporters, radiation shields, power supplies, and specialized communication and control systems.

The work scope for waste package retrieval will describe what comprises normal and off-normal retrieval activities, present an evaluation of the retrieval process, and develop strategies for normal and off-normal operation. This functional area will receive input from other disciplines, including electrical, structural, remote handling, ventilation, and radiological safety. Work in this area will also support conceptual evaluations of retrievability associated with the alternative design evaluations to support licensability criteria used to evaluate the alternatives.

This design will provide input to the systematic identification and evaluation of design basis events to classify systems, structures, and components based on safety.

In turn, the efforts in this area will contribute significantly to the development of a preclosure safety case by outlining the feasibility of retrieval using demonstrated technology and conforming to accepted design criteria. The work scope will include the results of other work that describes disruptive processes and events, and develops deliverables (drawings, outline specifications, and analyses) that reflect that input. The resultant information may also be used in other design efforts and used to support evaluation of alternative designs to ensure that the waste packages can be retrieved over the time required.

This work package will provide information to resolve the key technical issue of Repository Design and Thermal-Mechanical Effects, specifically for the subissue of retrievability. The design will be judged for its ability to demonstrate functionality and feasibility. Consequently, this design is considered vital information in reviewing and judging the adequacy of the LA.

3.2.1.7 Performance Confirmation

Work in this area will support Milestone M2JG presented in Section 7. In general, the goal of performance confirmation is to validate the results of the models used to estimate how the repository would act under postulated conditions (see Volume 3). The focus of this functional area will be to refine the layout of the repository and the configuration of the emplacement drifts, and update the plans for installing and designing special instrumentation and equipment to monitor "in-drift" and "out-of-drift" conditions. In-drift means inside the emplacement drifts containing the waste packages; out-of-drift means within the rock mass itself and in other underground areas. This design task addresses a system that is designated Bin 3 as follows:

- Parameters and criteria will be refined to further develop a strategy and associated processes and equipment for documenting and confirming repository performance.

- Requirements and assumptions will be selected from applicable documents and combined with other design bases to develop the design. For example, information from subsurface layout, ventilation schemes, and available utilities will be included.

- Performance confirmation concepts for alternative designs will also be developed to support cost and licensability criteria during the selection process for the LA design.

The locations for permanently installed monitoring stations will be estimated in the subsurface layout, and requirements for utilities and access will be factored into the selected subsurface design. This task will also evaluate and describe contingency activities related to confirming repository performance.

This work scope will develop information that supports the preclosure safety case by providing input to the systematic safety classification of design items that lead to identifying design basis events. Additionally, the systems used to gather data to help refine the predictions for postclosure performance can also be used to support preclosure safety by providing additional defense in depth monitoring capability for low-probability events (such as waste package breach). This work will also demonstrate that the project has the ability to confirm the performance of the repository during the retrieval period. Evidence of this capability will be shown in designs for reliably functioning systems and components for an extended period of time, and under potentially unfavorable conditions. By demonstrating that a desired set of data can be reliably acquired over a given period, predictions may be made as to the future performance of the repository, relative to predictive models.

The information will support the postclosure safety case by demonstrating how data will be supplied to future evaluations that will estimate expected performance and by aiding in interpreting relevant natural analogs. The nature of this work is largely confirmatory, and the methods used for continued evaluation and data gathering will be shown to be reliable and feasible for long-term application.

This work package will address the key technical issue dealing with Repository Design and Thermal-Mechanical Effects, specifically in the area of drift stability. Information about drift stability will feed directly into the evaluation of thermal loading, retrievability, and potentially disruptive fault structures. The work package will also provide input to two other key technical issues, Structural Deformation and Seismicity and Evolution of the Near-Field Environment, by considering seismic motion in the design and the impact of mechanical and chemical changes in materials used for ground support in the emplacement drifts. This work will also address aspects of the NRC key technical issues on Repository Design and Thermal-Mechanical Effects, Evolution of the Near-Field Environment, and Thermal Effects on Flow.

3.2.1.8 Sealing and Closure

Work in this area will support Milestones M2MX and M2MV presented in Section 7. This task will assess requirements and develop preliminary designs for closing the various openings of the repository, including sealing and backfilling underground openings, and addresses a system that is designated Bin 3. The sealing and closure process will involve two, and sometimes three, steps. First, structures previously installed during the characterization and developmental phases of the project must be removed in accesses and most underground areas. Then, the seals must be installed. The seals for ramps, shafts, and drift areas will be site specific and will be specially designed, with particular attention to horizontal and vertical applications. Finally, some of the areas will have to be backfilled. In many cases, the sealing and backfilling processes will be sequential, that is, repeated in succession. The materials and construction techniques to be used will be selected to enhance the longevity of the final seal. Currently, the work in this area would not include the design of the backfill for the emplacement drifts. Information already exists in this area and may be enhanced pending the outcome of alternative design efforts and performance assessment modeling. Should the backfill feature be selected as part of the design being developed for the LA, the design will specify the types of material to be used and the requirements for storing, and if necessary, blending them; moving them underground; and installing them (backfill emplacement). The structural components required for these operations will also be specified.

The equipment for backfilling will be selected based on suitability and ease of use. The design for backfilling the shaft collar and ramp portal area will make closing the facility and reclaiming the site area easier.

The design activity will provide input to modeling that will estimate expected performance and aid in evaluating techniques to reduce the amount of water that could enter the drifts and drip on the waste packages. The work in this area could affect the decision on ventilating the emplacement drifts longer than currently planned, to dry out the host rock and reduce the humidity for as long as possible. Currently, the confidence in the overall effect of ventilation is low. However, raising this confidence level to moderate, could lead to selecting long-term ventilation as a feasible design. To a lesser extent, the results of this work will contribute to the postclosure safety case by reducing the intrusion of water through the shafts and ramps. This information will be used to calculate margins of safety and evaluate defense in depth. Installing properly designed seals and applying specific closure techniques will help effectively block human intrusion through openings used to access the repository.

This work package will help resolve the key technical issue of Repository Design and Thermal-Mechanical Effects, specifically in the area of drift stability. The package will also address a second key technical issue, Evolution of the Near-Field Environment, by considering seismic motion in the design and the impact of mechanical and chemical changes in materials used for ground support in the emplacement drifts.

3.2.1.9 Subsurface Utilities

Work in this area will support Milestone M2MV presented in Section 7. This work scope will address all the subsurface utilities required during developmental and operational phases of the project. Though each phase will require different types of equipment and levels of system functions, every effort will be made to integrate systems based on safety and economy.

This task will involve developing general descriptions of subsurface utilities, such as the supply and distribution of the following:

- Water and power
- Lighting
- Compressed air
- Communications (telephones and intercom systems)
- Fire protection and emergency warning systems
- Methods to remove excess water used in certain processes

Except for lighting, which is designated Bin 1, each of these utilities is classified as Bin 2. Little definitive work has been done for these systems to date. The bulk of the initial design work will be performed during this phase of the project.

This design will provide input to the systematic identification and evaluation of design basis events to classify systems, structures, and components based on safety.

As such, this work will provide input to the preclosure safety case as it relates to worker safety and the reliability of systems related to radiological safety. The information will not significantly support the postclosure safety case because all utilities will be removed from the subsurface before the repository is sealed and closed. However, a design basis event that compromises the integrity of the water system could cause the unexpected introduction of excess water to the host rock. Hence, the assurance of a properly designed system to mitigate this potential is essential in the licensing process as a demonstration of defense in depth or margin of safety.

The subsurface utilities work scope will help resolve the key technical issue of Repository Design and Thermal-Mechanical Effects by providing information on systems that support other key systems. Virtually every other system requires some degree of utility interface to work properly and be maintained. The reliability, availability, and maintainability of individual systems and repository processes depend on a well-designed, dependable, global utility system.

3.2.1.10 Subsurface Integrated Control Systems

Work in this area will support Milestones M2MQ, M2MX, and M2MV presented in Section 7. This effort will involve designing components that allow some repository operations to be controlled remotely and is applicable to several of the alternative design evaluations. Such activities include emplacing and retrieving the waste packages and conducting certain performance confirmation activities. The work scope will include reviewing and refining important elements of existing remote handling and control design concepts. This review covers remotely operated vehicles, primary and secondary power distribution for remote systems, and mobile communication systems; identifies digital instrumentation and control components; and evaluates the long-term reliability and maintainability of the instrumentation selected. Various other design concepts will be addressed and will help to support the comparative alternative design evaluations.

This task will include the following:

- Focus on developing an initial plan and process for producing the software elements of the repository systems. Project-level issues related to software development will be addressed. These issues include regulatory precedents within NRC, product assurance and quality assurance controls, reliability testing, life cycle operations, and maintenance. In addition, this work will provide for analyzing overall functional requirements for software-based systems, strategies for developing software, and architectures for real-time control. The work scope will cover investigating and recommending software development platforms, languages, tools, and operating systems and developing preliminary, functional flow diagrams for software.

- Help evaluate and demonstrate key technologies for possible use within the repository. Focused demonstrations will include control systems for remotely operated vehicles, power sources and distribution for mobile equipment, environmental and reliability tests of digital instrumentation and control equipment, and investigation of high-temperature electronics.

- Provide for designing, developing, fabricating, and testing a small-scale, fully operational, laboratory engineering model. The engineering model will be used to investigate, address, and demonstrate system-level integration issues and design interfaces.

The results from the technology demonstrations and engineering model will help in designing the initial, full-scale working prototypes of critical, remotely operated vehicles for any subsurface remote operations. These full-scale prototypes will be built and demonstrated in parallel with the anticipated 3-year, license approval process.

This design provides input to the systematic identification and evaluation of design basis events to classify systems, structures, and components based on safety.

These assessments, along with supporting rationale, will be formally documented as input to design criteria. Conformance of repository system design with regulatory guidance will ensure that the design is consistent with established regulatory precedents, where applicable, and will thereby facilitate the licensing process. The information does not support the postclosure safety case.

This work package will provide information to close the key technical issue of Repository Design and Thermal-Mechanical Effects, specifically in the areas of waste handling techniques, retrievability, and evaluations of reliability, availability and maintainability. Remote handling functions are essential to the functioning and reliability of certain system components in potentially hostile environments associated with several of the alternative design options.

3.2.1.11 Radiological Safety

Work in this area will support Milestones M2MQ, M2MU, M2MX, and M2MV presented in Section 7. This work scope is fairly broad and will cover all the components and systems related to controlling and monitoring radiological conditions in the subsurface operational areas. Tasks include refining designs, providing design analyses, and describing systems in sufficient detail to understand their functions and capabilities. In addition, outlines for specifications for system components will be supplied. Detailed specifications will also be provided for critical components related to radiological safety, monitoring, control, and all safety-related components and systems for radiological control.

Computational fluid dynamics calculations will be used to model the release, deposition, and dispersion of radioactive materials under various conditions. This modeling will assess the consequence of the release, transport, and deposition of radioactive materials in subsurface regions. These calculations will be made using a commercially available, computational, fluid dynamics code that will be verified and validated. Results from the analyses will be used in other engineering tasks to design such items as shielding, barriers, components for monitoring, and specialty items that must perform in radioactive environments. In addition, concepts associated with enhanced access design scenarios will be developed and evaluated. As a result, all selected designs impacting radiological safety will be analyzed and ultimately optimized as much as possible, to achieve radiation protection that is as low as reasonably achievable.

The radiological safety work scope will develop information that will support the preclosure safety case by using demonstrated technology and accepted criteria and provides input to the systematic identification and evaluation of design basis events to classify systems, structures, and components based on safety.

As design basis events are developed, the work in this area will integrate the results with designs that are developed for preclosure activities. In this respect, aspects of the preclosure safety case will be continuously addressed and updated as new information is developed relative to the preclosure safety strategy as discussed in Section 2.3. The information will not support the postclosure safety case directly but will contribute to the design of systems and components that will be used in sealing and closing the facility.

This work package will provide information to resolve the key technical issue of Repository Design and Thermal-Mechanical Effects, specifically in the areas of worker safety and compliance with standards set by NRC.

3.2.1.12 Engineered Barrier System Performance Modeling and Testing

Work in this area will support Milestones M2MQ, M2MU, M2MX, and M2MV presented in Section 7. This work scope will address how certain features of the engineered barrier system, exclusive of the waste package, perform. The work will be specifically tailored to meet the needs described in Section 2, particularly the outcome of the performance allocation process described in Section 2.2.6.

As noted in Section 2.2.4, the goals for increased confidence in assessment of postclosure performance affect the engineered barrier in several aspects. One of the principal factors assigned relatively high priority is the dripping of water onto the waste packages (see Table 2-2). Various options including drip shields, backfill, and Richards' barriers have the potential to mitigate the quantity of water that may drip onto waste packages. The work will provide both predictive modeling and experiments. The engineered barriers will be tested in laboratory settings, as well as in larger proof of principle testing arrangements. Predictive models will be confirmed by the tests. It is expected that design options to control water dripping onto the waste packages will significantly enhance expected performance, defense in depth, and other aspects of the postclosure safety case. Additional work will be undertaken to better define the potential of the waste package line loading concept to reduce the quantity of water that may contact the waste packages.

An additional aspect of this work concerns the review and testing of chemical and microbial effects of materials that potentially comprise the engineered barrier system design, including concrete and steel. These materials may have both positive and negative effects on corrosion of the waste package and radionuclide transport or sorption capability of the unsaturated zone.

Potential backfill material will also be mechanically tested. The backfill, if used, will serve to protect both the waste packages and the drip shields. Mechanical testing will support accurate prediction of stress conditions for key barriers, resulting from various rockfall conditions.

Materials for the engineered barrier system will be reviewed and their source specifications cataloged and qualified. Various natural and long-lived man-made analogs to the engineered barrier system will be investigated and their applicability to verifying performance of the repository will be determined. The VA reference design and alternative designs will be evaluated using the models and testing associated with this effort.

This work will address two key technical issues, Effects on the Near-Field Environment and Thermal Effects on Flow. It will also address questions and concerns from the Nuclear Waste Technical Review Board and the Repository Consulting Board. Products of the work will support the repository subsurface design, waste package design, and performance assessment and will be incorporated into the LA either directly or as backup technical documentation.

3.2.2 Waste Package

This section presents the work planned for waste package design between the time this VA is issued and the LA is submitted. The key milestones and the schedule associated with the activities discussed in this section are provided in Section 7.

The design analyses and testing programs for the waste packages have been planned to resolve issues that impact preclosure operations and postclosure performance identified in Section 2. The work scope for the waste package focuses on analyses and tests that reduce remaining uncertainties for the principal factors affecting the long-term performance of a repository. Planned work also addresses the ability of the waste packages to provide defense in depth and to minimize the impact of disruptive events on their performance. The analyses further support waste package designs that provide containment for design basis events that could occur during the preclosure period. Additionally, more fundamental changes to the waste package design are being revisited, evaluated, and documented to support a quantitative evaluation of design alternatives and performance enhancement features necessary to support the initial selection of an LA design. The design work addresses aspects of several of the key technical issues identified by NRC (Section 4.3.3).

The work scope is organized into nine functional areas as follows:

- Waste package designs for uncanistered spent nuclear fuel
- Waste package designs for canistered spent nuclear fuel
- Waste package designs for DOE-owned spent nuclear fuel
- Waste package designs for high-level radioactive waste
- Alternate waste package designs
- Disposal container fabrication and closure welding
- Criticality methodology
- Waste form testing and modeling
- Waste package materials testing and modeling

Table 3-7 shows the relationship between each of these areas, the preclosure and postclosure safety cases, and the references to cost and schedule information.

The first five functional areas deal with designing long-term waste packages to accept various types of highly radioactive materials that may arrive at the proposed repository in different forms and/or conditions. As indicated in Volume 2, Section 5.1.2, several variations of waste package design configurations are necessary to accommodate the range of fuel types and waste forms. Many of the engineering analyses for these five areas are similar, as are the databases that must be established and monitored. These redundancies will be noted with a summary statement, and the specific details will not be repeated.

Analyses for waste package designs for the LA will be to the level of detail necessary to address preclosure operations and postclosure performance. Though work on all designs will continue, it is currently anticipated that six designs will be discussed in the LA as required to demonstrate compliance for all waste package designs and all waste forms. After the LA has been submitted, all the designs will be brought to the level of detail required for procurement and fabrication.

The last four functional areas deal with testing and/or analyses to support the design activity. This work supports and/or validates the waste package designs, helps evaluate the feasibility of the design and its expected performance, and defines the criticality methodology that will be applied to demonstrate compliance with 10 CFR 60.131.

All of the design work and testing programs for the nine functional areas support the same principal factors for postclosure performance and preclosure operations. Collectively, they also address aspects of the NRC key technical issues, and therefore, will be discussed here as a whole, rather than in each individual section.

Postclosure Performance. As discussed in Section 2.2, the principal factors affecting the postclosure performance of the waste package include:

- Water dripping onto the waste package
- Humidity and temperature at the waste package
- Chemistry of water on the waste package
- Integrity of the carbon steel outer barrier of the waste package

Table 3-7. Application of Planned Waste Package Design Work to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Post-closure Safety Case[1]	Elements of the Postclosure Safety Case[1]: 1. Expected Postclosure Performance — Principal Factors Supported by Technical Work	Elements of the Postclosure Safety Case[1]: 1. Expected Postclosure Performance — Priority to SR/LA[2]	2. Design Margin/ Defense In Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Performance Confirmation	Preclosure Safety Case[3]	PSS Activities[4]
Waste Package Designs for Uncanistered Spent Nuclear Fuel (3.2.2.1)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√				√	2371 2375 2377 2390 6113
Waste Package Designs for Canistered Spent Nuclear Fuel (3.2.2.2)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√				√	2371 2375 2377 2390 6113
Waste Package Designs for DOE Spent Nuclear Fuel (3.2.2.3)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√				√	2371 2375 2377 2390 6113
Waste Package Designs for High-Level Radioactive Waste (3.2.2.4)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√				√	2371 2375 2377 2390 6113
Alternate Waste Package Design (3.2.2.5)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√				√	2021 2377 2391
Disposal Container Fabrication and Closure Welding (3.2.2.6)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√	√			√	2373
Criticality Methodology (3.2.2.7)	√	Transport through and out of engineered barrier system Flow and transport in the unsaturated zone	1 3	√				√	2380
Waste Form Testing and Modeling (3.2.2.8)	√	Integrity of spent nuclear fuel cladding Dissolution of spent nuclear fuel and glass waste forms	2 1	√				√	7030 7032
Waste Package Materials Testing and Modeling (3.2.2.9)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier	1 3	√				√	7040 7042

[1]Postclosure safety case described in Section 2.2

[2]Postclosure work prioritization for site recommendation and LA described in Section 2.2.4

[3]Preclosure safety base described in Section 2.3

[4]Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule in Sections 6 and 7

- Integrity of the high-nickel alloy inner barrier of the waste package

- Integrity of spent nuclear fuel cladding

- Dissolution of the uranium oxide and glass waste forms

With the exception of the criticality methodology, the core of the work in the nine functional areas addresses how the principal factors affect the lifetime of the waste packages and their ability to retard the migration of radionuclides. This information is obtained by using data generated in the testing programs to aid in selecting materials that perform well under the anticipated repository conditions. Once the materials are selected, design analyses are required to evaluate waste package structural integrity, thermal performance, and shielding properties. In addition, the testing programs also study the containment provided by the waste package barriers and the spent nuclear fuel cladding. Although most of the tests are in progress, many are long term and will continue up to, and beyond, submittal of the LA.

The criticality analysis work provides data to help address the principal factor dealing with the transport of radionuclides through, and out of, the waste package. The amount of radioactive material available for transport is a factor in determining the rate at which radionuclides will move away from breached waste packages. The increase of radionuclide inventory caused by an unlikely criticality event is estimated through specific postclosure criticality analyses.

The waste package design also supports potential options and alternatives that could reduce uncertainties and increase confidence in the postclosure performance. Work on waste package alternatives is presented in Section 3.2.2.5.

Preclosure Operations. It has been determined that the waste package has functions related to radiological safety and waste isolation that have no applicable regulatory precedent. As a result, the waste package has been designated a Bin 3 item. The designs for structures, systems, and components designated as Bin 3 require the most detail and the design analyses must include evaluations to demonstrate compliance with the design basis events. The analyses performed for the waste package designs include demonstrating compliance for the design bases events that will support the LA.

NRC Key Technical Issues. The work scope in this section addresses various aspects of six of the NRC key technical issues: Structural Deformation and Seismicity (Section 4.3.3.2), Near-Field Environment (Section 4.3.3.3), Container Life and Source Term (Section 4.3.3.4), Repository Design and Thermal-Mechanical Effects (Section 4.3.3.6), Total System Performance Assessment and Integration (Section 4.3.3.7), and Radionuclide Transport (Section 4.3.3.10).

- **Structural Deformation and Seismicity.** The waste package designs must be able to withstand structural deformation caused by seismicity and related hazards to ensure that any damage would not pose a risk of noncompliance.

- **Near-Field Environment.** The waste package lifetime is influenced by the amount of water contacting the waste package and the chemistry of the water. The thermal loading must be considered for its effect on the waste and the natural environment.

- **Container Life and Source Term.** The objective of the waste package analysis work and the materials tests is to design a long-lived waste package. Work being done includes evaluating how to extrapolate long-term performance from short-term laboratory data, determining how long-term exposure to high temperatures and mechanical loads affects the mechanical stability of waste package materials, assessing how microbiological organisms affect the performance of waste package materials, and selecting waste package materials that will provide high performance in the anticipated repository environment. In addition, waste form testing supports the evaluation of factors that affect what products altered waste forms produce and how they impact the release of radionuclides. The radionuclide inventory from the consequence of a criticality is also studied to determine the source term.

- **Repository Design and Thermal-Mechanical Effects**. One of the subissues associated with this NRC key technical issue is demonstrating that the designs for the repository and the waste packages meet the preclosure and postclosure performance objectives. Analyses will be performed to support these objectives for the waste package design. A rational basis for allocation of performance to the waste package is anticipated as a result of evaluating the design features and alternatives.

- **Total System Performance Assessment and Integration**. A total system performance assessment will be conducted to determine compliance with limits set for dose and risk. Criticality analyses will be performed to address the disruptive event of a postclosure criticality. The tests being done on the waste package materials and waste forms help determine degradation modes under repository conditions. This information, in turn, feeds into the total system performance assessment to project offsite dose limits.

- **Radionuclide Transport.** This key technical issue is concerned with processes that may affect the movement of radionuclides away from the repository toward the accessible environment. The waste package materials and waste form tests contribute to understanding these processes.

3.2.2.1 Waste Package Designs for Uncanistered Spent Nuclear Fuel

Spent nuclear fuel assemblies will come from both pressurized water and boiling water commercial nuclear power reactors and will vary dimensionally, and in enrichment and burnup. Collectively, they will represent all of the various reactor designs that have operated and discharged spent nuclear fuel. Under current operational scenarios, this spent nuclear fuel will be transported to the proposed repository in transportation casks that are not suited for long-term disposal. Some of these casks will contain individual fuel assemblies, while others will contain fuel assemblies that are already in sealed canisters. Fuel that arrives as individual, intact assemblies is referred to as "uncanistered fuel." This "uncanistered fuel" will be removed from the transportation cask and placed directly inside the disposal container.

The design process incorporates several steps as outlined below:

- Identify the different types of spent nuclear fuel from commercial reactors. These fuel types include damaged and undamaged (intact) fuel and fuel whose configuration has been changed in some way (for example, consolidated).

- Determine off-normal events and accidents that could occur (design basis events).

- Perform thermal, structural, criticality, and shielding analyses on the waste package design and design alternatives.

- Prepare documents that describe the waste package designs and design alternatives, and provide necessary information to other project organizations.

Eight different waste packages are being considered to accommodate the varying criticality and thermal parameters associated with the uncanistered commercial spent nuclear fuel: five for fuel from pressurized water reactors; three for boiling water reactor fuel. Engineering analyses will be conducted to evaluate each design, as well as the alternative design concepts, as they are applied and as they interact with other elements and features in the underground environment. These analyses will include thermal, structural, criticality, shielding, and ionizing radiation evaluations. They will incorporate current data from tests of the materials comprising the waste packages and the waste forms they contain.

Developing input to these analyses requires calculating and updating the temperature conditions inside and outside the waste package. The structural analyses will consider normal conditions (e.g., static loads, thermal loading, and handling loads), as well as those for design basis events (e.g., drops, tipover events, rockfalls, and transporter accidents). The analyses that address shielding and the effects of ionizing radiation will evaluate the extent of radiation-induced corrosion, the amount of shielding required to support the welding operations that permanently seal the containers, and the amount of shielding and impacts to waste package configurations for self-shielding alternatives. These evaluations will help determine the thickness of the materials comprising the waste package. Internal component designs (thickness, configurations, and potential additives and fillers) will be examined by thermal, structural, and criticality analyses.

Criticality evaluations will be performed for preclosure operations and postclosure performance. In addition, numerical plots will be developed to define the acceptable range of characteristics of fuel to be loaded into the various disposal containers. These plots consider the fuel's initial concentration of uranium-235 (enrichment) and the time and power at which the fuel operated (burnup). The plots, called loading curves, will be generated for waste packages in two states: waste packages that are intact as well as those that have degraded.

Mixed-oxide fuels that can be used to burn plutonium in existing commercial power reactors are also being studied. Mixed-oxide spent nuclear fuel has higher temperatures and more residual radioactivity than normal commercial spent nuclear fuel. It therefore requires additional analyses to ensure that it is compatible with the waste package design for commercial fuel. Since mixed-oxide fuel designs have not yet been licensed in the United States, the conceptual waste package design is based on representative data of mixed-oxide fuel used elsewhere. If the actual data for the mixed-oxide fuel should differ, the design may change. These analyses will lead to additional work to support the LA.

The waste package design for the uncanistered spent nuclear fuel work scope supports Milestone M2MW, Final LA Design & Verification, presented in Section 7.

3.2.2.2 Waste Package Designs for Canistered Spent Nuclear Fuel

Spent nuclear fuel that arrives at the repository already encased in canisters is referred to as canistered fuel. Canistered fuel will arrive at the repository in transportation casks (overpacks) that contain sealed canisters which must be demonstrated to be suitable for direct, long-term disposal in the waste package. The canisters containing the fuel will be removed from the transportation casks and placed into the disposal containers. Numerous vendors could supply these canisters, which could be different for different fuel types. For the canisters to be considered suitable for long-term disposal, they must be designed, fabricated, and loaded with spent nuclear fuel based on specifications and criteria approved by DOE and NRC. As a result, the work scope of this section will include reviewing the different canister designs and maintaining a database of those approved by NRC.

Engineering analyses similar to those performed for the uncanistered fuel will be performed to evaluate the acceptability of the disposal-related components for the different types of canisters.

The waste package design for the canistered spent nuclear fuel work scope supports Milestone M2MW, Final LA Design & Verification, presented in Section 7.

3.2.2.3 Waste Package Designs for DOE-Owned Spent Nuclear Fuel

This work scope covers designing the waste packages for DOE-owned spent nuclear fuel, which comes from sources other than commercial reactors. DOE currently has more than 250 types of spent nuclear fuel in its inventory, including spent nuclear fuel from the Naval Nuclear Propulsion

Program. A significant fraction of this material is metallic uranium based fuel from weapons production reactors and poses 10 CFR 60.135 chemical reactivity issues that are not concerns for commercial fuel and high-level radioactive waste glass. As in the case of canistered commercial fuel, this fuel will arrive at the repository in canisters that must be demonstrated as suitable for direct, long-term disposal in the waste package. It is likely to be less well characterized and more damaged than the bare commercial fuel. Again, the canister developed at the custodian site may not necessarily have been designed specifically to conform to the requirements of long-term storage in a waste package.

Several organizations are cooperating in this effort. Collectively, their work will ensure that DOE-owned spent nuclear fuel sent to the repository will have acceptable characteristics for emplacement and will meet NRC requirements for disposal.

After DOE-owned spent nuclear fuel has been evaluated and characterized, conceptual designs will be developed and initial designs will then be optimized, as needed. Thermal analyses will be done to evaluate the thermal performance of the canister and the waste package designs under steady-state and normal conditions. Peak temperatures inside the waste package will also be calculated. Structural, criticality, and shielding analyses (as described in previous sections) will also be performed for the canister and waste package.

Several activities are being conducted through the National Spent Nuclear Fuel Program at the Idaho National Engineering and Environmental Laboratory. The work is organized into six tasks that include the following:

- Preparing a document that identifies the data required for each type of DOE-owned spent nuclear fuel and ensures that the proposed repository can accommodate the specific fuel type
- Developing an interface control document that will serve as the vehicle for documenting and controlling agreed upon design solutions
- Evaluating the impact of DOE-owned spent nuclear fuel on the overall performance assessment of the repository and identifying the sensitive parameters
- Evaluating the potentially important characteristics of this fuel with respect to 10 CFR 60.135 requirements (e.g., chemical reactivity, pyrophoricity, particulate material, gas generation, and residual water content) to determine their impact on repository performance and establish appropriate guidelines
- Evaluating consequences of design basis events
- Analyzing the criticality safety margin for the representative types of DOE-owned spent nuclear fuel in both their intact and degraded postclosure configurations

The work described above is in progress and will identify issues requiring follow-on work in the years between submittal of the VA and submittal of the LA. All of the criticality analyses will be finished in 2 to 3 years. However, as each of the representative types of spent nuclear fuel is analyzed, the need for additional analyses to support the LA is likely.

The waste package design for the DOE-owned spent nuclear fuel work scope supports Milestone M2MW, Final LA Design & Verification, presented in Section 7.

Waste Package Design for Spent Naval Fuel. Spent nuclear fuel from the U.S. Navy will be delivered to the repository in canisters. It is assumed that these canisters will have been demonstrated as suitable for direct, long-term disposal in the waste package. Each disposal container will hold a single canister. Because this fuel design is classified, the Office of Naval Reactors will perform detailed analyses of intact naval spent nuclear fuel. DOE will then review these analyses. Parameters will be presented in a form that will avoid disclosing classified information and restricted data or information important to national security and will be submitted in a separate appendix to the LA, if necessary.

This cooperative effort will include several tasks. First, the data and analyses on the criticality safety of the canisters loaded with the spent naval nuclear fuel will be reviewed, as will the degradation scenarios and configurations of the fuel and the materials for controlling criticality after the canisters have been breached. DOE will analyze fuel that is no longer intact for criticality. The Office of Naval Reactors will provide the source term for the release of material from the canister once it has been breached. This source term will be incorporated into the DOE model for analyzing both near-field and far-field criticality for all spent nuclear fuel releases in the repository and the release-rate model for performance assessment of the repository.

The waste package design for the spent naval nuclear fuel work scope supports Milestone M2MW, Final LA Design & Verification, presented in Section 7.

3.2.2.4 Waste Package Designs for High-Level Radioactive Waste

In addition to the commercial spent nuclear fuel or DOE-owned spent nuclear fuel, the repository will also contain high-level radioactive waste generated from processing spent nuclear fuel to recover uranium and plutonium. This reprocessed fuel comes from various types of reactors: weapons production, research and test, naval propulsion, and commercial power. DOE will process this high-level radioactive waste into a solidified, insoluble (vitrified) glass or ceramic. The solidification process will trap the radioactive fission products in stabilizing matrices such as borosilicate glass. The solidification will be done at four locations: the Savannah River and the Hanford sites, the Idaho National Engineering and Environmental Laboratory, and the West Valley Demonstration Project.

This work scope covers the waste package designs for the canisters of the solidified, high-level radioactive waste that will arrive at the proposed repository. These canisters will be placed directly into the long-lived disposal containers.

Several waste package designs will be required for the canisters with vitrified, high-level radioactive waste. The "standard" waste package design will accommodate canisters from the Savannah River site or from other sites that use the same size canisters. Additional designs, called co-disposal options, will accept a DOE-owned spent nuclear fuel canister in addition to the vitrified, high-level radioactive waste canisters. The number of designs required will be based on high-level radioactive waste and DOE-owned spent nuclear fuel canister designs currently being developed. As specific DOE-owned spent nuclear fuel canisters and loadings are identified, the design for the co-disposal waste packages will be evaluated to ensure that all the criteria are met.

As with uncanistered fuel, a waste package database of the different canister designs being developed will be maintained and the designs will be reviewed. In addition, the requirements for placing the canisters inside the disposal containers will be developed and issued. The engineering analyses to evaluate the acceptability of the high-level radioactive waste disposal container and its components are the same as those previously described for canistered fuel.

Immobilized Plutonium. The work on plutonium disposition will evaluate how optional forms of surplus plutonium from the weapons program can affect the proposed repository. Several different mixed-oxide fuel designs and immobilized plutonium (glass and ceramic) have been evaluated. The principal effort to support the LA will be evaluating the long-term potential for criticality of the most recent mixed-oxide fuel design and the latest ceramic waste form for immobilization. These analyses will include assessing likely scenarios for degradation of the waste package (including the waste form) and the criticality of the resulting configurations of the degraded material, and estimating the probability of any scenarios leading to a critical configuration. Assessing the potential for critical configurations will include locations both inside and outside the waste package.

The waste package design for the high-level radioactive waste work scope supports Milestone M2MW, Final LA Design & Verification, presented in Section 7.

3.2.2.5 Alternate Waste Package Designs

In addition to the activities necessary to support evaluating the design alternatives and subsequently selecting an initial LA design, several specific concepts for alternative waste package designs will be developed to further satisfy the requirements of 10 CFR 60.21. Optional design features will be based on proven performance and will be investigated for their effect on improving the overall performance of both the engineered barrier system and the repository site or their contribution to defense in depth.

As a part of the design process, and in compliance with regulatory requirements, a comprehensive quantitative evaluation of design features and design alternatives will be completed to support selecting the initial LA design. A comprehensive discussion of design features and alternatives can be found in Volume 2, Section 8. Existing pertinent information previously gathered for the targeted features or alternatives will be used and summarized as much as possible. However, additional near-term efforts are anticipated to quantitatively compare advantages and disadvantages associated with given features or alternative designs. A predetermined set of ranking criteria will be used to support the comparisons, from which an initial selection of the LA design will be made. These considerations emphasize the feasibility of the concepts, their implications as far as radiological safety and cost, and other information pertinent to the selected evaluation criteria. Performance implications associated with the features and alternatives will generally be determined by performance assessment calculations. Waste package design efforts will focus on several alternative categories and design features, including the following:

Category 1. Containment Within the Engineered Barrier System

- Waste package materials
  - One or two corrosion-resistant materials
  - Ceramics–options for application to waste packages and drip shields
- Barriers
  - Drip shield options
  - Backfills, Richards' barrier–material compatibility, thermal requirements
  - Diffusive barrier or getter under waste package
- Internals
  - Canisterized assemblies
  - Additives and fillers
- Emplacement modes

Category 2. Other Engineered Enhancements

- Metal lined drifts
- Unlined drifts–effects of rockfall on the waste package (also applicable to deferred closure)
- Ground support materials–material compatibility

Category 3. Integrated Effects of Thermal Loading

- Waste package size
- Thermal loading and thermal management (e.g., pre-emplacement aging, blending, spacing, line loading)
- Ventilation
- Rod consolidation

Category 4. Waste Package Production and Emplacement Operations

- Waste package closure and shield thickness
- Waste package fabrication process
- Self-shielding

Efforts will be focused on developing concepts and obtaining performance (corrosion models), cost, schedule, and other information pertinent to the alternative evaluation criteria. The results of this effort may alter the reference design approach, to be followed by more detailed design analyses necessary to support the site recommendation and the LA.

The alternate waste package design work scope supports Milestone M2MQ, Select Initial SR/LA Design & Options, presented in Section 7.

3.2.2.6 Disposal Container Fabrication and Closure Welding

Drawings and parts lists for each type of disposal container will be continually updated during the design process. Final fabrication drawings will be prepared after the LA has been submitted.

The fabrication techniques for the disposal containers will be selected and refined, and will comply with the applicable standards or codes. Mockups will be built to test the fabrication techniques and for proof of principle testing. Engineering sketches and specifications for these mockups will be prepared as required.

In addition, the welding process for sealing the disposal containers will be developed and refined using mockup blocks that are fabricated under the same process as that for the actual disposal containers. All welds will be measured for residual stress. This information will be used for design engineering and laboratory evaluation. Samples from these mockups will be evaluated in the laboratory.

Remotely operated equipment will inspect the closure welds and the disposal containers. These systems will be developed and the design requirements and procurement documents prepared.

The types of material used for the disposal containers will influence the fabrication method and welding process. As different materials are considered during the design process, additional fabrication and mockups will be needed. Additionally, self-shielding options and various waste package size alternatives to be evaluated in the near term could alter the fabrication and closure inspection techniques. Closure feasibility and subsequent inspection techniques will need to be demonstrated to support the LA, particularly considering the importance placed on the closure process for both the preclosure and postclosure safety arguments.

The disposal container fabrication and closure welding work scope supports Milestone M2MW, Final LA Design & Verification, presented in Section 7.

3.2.2.7 Criticality Methodology

A topical report on the methodology for analyzing disposal criticality will present the comprehensive method for determining how to analyze the potential for criticality during the postclosure phase of the repository (including the use of burnup credit) for NRC acceptance. After NRC has accepted the method, it will be used in the LA to demonstrate the acceptability of the proposed criticality safety system. The topical report will be issued for NRC review in October 1998. Though this initial submittal focuses on commercial spent nuclear fuel, a partial example of how the method can be applied to DOE aluminum clad fuel will also be included. Amendments will be developed to incorporate other DOE and U.S. Navy waste forms, as well as additional chemical assay and commercial reactor critical data.

The topical report has two major subject areas: the basis for the methodology, and the application of burnup credit. The first subject area, the risk-informed methodology, covers selecting and building possible scenarios and configurations that could affect every waste form and every waste package design, evaluating the probability of their contributing to criticality, and estimating the consequences of a criticality. Additional test data may be required for each new or unique waste form for intact, in-package degraded, and external degraded configurations.

Work to support the first subject area concentrates on improving the methodology. This work includes determining the range of parameters to describe the potentially critical configurations; refining the method for generating probabilistic distributions of parameters affecting criticality (e.g., the fraction of emplaced spent nuclear fuel that may become critical); and although a criticality is unlikely, estimating the consequences for the probable criticalities (e.g., peak power, total energy, and increase in radionuclide inventory).

The second area presents how burnup credit is applied. The use of burnup credit is a major distinction from other typical methods that are presently used for NRC work. This work scope will include determining the use of a measurement device for verifying records of assembly burnup and measuring how the power was distributed through the assembly (axial profile). NRC is expected to impose a requirement for using a measurement device for waste package designs using burnup credit.

A preclosure criticality process will be developed to supplement the postclosure method described in the topical report. The process will include the burnup credit method and steps for meeting 10 CFR 60.131.

The criticality methodology work scope supports Milestone M2FV, Disposal Criticality Analysis Methodology Topical Report, presented in Section 7.

3.2.2.8 Waste Form Testing and Modeling

Waste form materials will be tested to help develop appropriate models to simulate how the materials could be altered and oxidized, and how they would release radionuclides. These tests include long-term oxidation, dissolution, surface alteration, and radionuclide release rates for spent nuclear fuel (including high burnup commercial fuel and representative DOE-owned spent nuclear fuels) under a range of wetting modes. Similar tests will be conducted for glass-type waste forms.

The specific activities planned to support the LA fall into four categories. The first three relate to spent nuclear fuel and cladding, while the fourth deals with a borosilicate glass form.

- Long-term testing of waste forms under dripping conditions
- Oxidation tests using thermogravimetric analysis
- Flow through dissolution, including tests on cladding and hardware
- Tests on borosilicate glass

Spent nuclear fuel and high-level radioactive waste glass will be tested under conditions anticipated in the repository. These tests will provide data on dissolution and release rates for film flow and dripping water conditions. In one test, spent nuclear fuel, both in cladding and crushed in a thin film, will be tested for cladding integrity, bounding alteration, and release rates. In a separate test, colloids that form during the degradation of spent nuclear fuel and glass samples will be analyzed. Another test assesses how dripping solutions interact with potential waste package and emplacement drift materials such as crushed tuff, concrete, and corroded metal products.

Degraded glass-type waste forms will continue to be tested semiannually with actinide-doped material coming from the Defense Waste Processing Facility and the West Valley Demonstration Project. Parametric tests will continue including evaluating surface area, waste chemistry, and temperature. The tests will be adapted to assess interactions between leachates and waste package component materials. The reaction rate for glasses of the Defense Waste Processing Facility type with water vapor will be determined as a function of temperature.

Oxidation will be measured using thermogravimetric analysis techniques, which quantify changes in sample weight as a function of time and temperature. These tests will help in understanding, quantifying, and modeling how uranium dioxide fuel oxidizes (kinetics).

Dissolution rates will be obtained from flow-through dissolution tests on spent fuel samples under controlled chemistry and temperature conditions. The data will be used in developing models and as input for analyzing and interpreting results from the unsaturated drip testing. In addition, the bounds for release rates of highly soluble fission products will be obtained.

A preliminary review of models for cladding degradation has identified the need for data on specific parameters of the model. The testing program for most specific parameters will provide these parameters via short-term and long-term testing of cladding sections. In addition, a model for the failure of Zircaloy cladding will be developed. Several mechanistically different cladding failure modes will be evaluated for a minimum lifetime-to-failure model. This, in turn, will support more definitive allocation of postclosure performance requirements to the fuel cladding, as well as show compliance with the allocation.

The release of radionuclides from spent nuclear fuel hardware may also be measured. Inconel and stainless steel components contain small amounts of long-lived radionuclides such as nickel-59. Simple static leach tests will be conducted in well water for varying periods of time. Initial radionuclide analyses of the test solutions will be conducted.

The current model for dissolving a glass-type (borosilicate) waste form has some uncertainties. This model will be improved by incorporating additional test data and production information from the glass producers. Experimental data sets will be used to evaluate model parameters from various tests on glass waste forms.

Borosilicate waste glass will be exposed to flow-through tests, closed-system tests, and surface-titration tests over a range of pHs (acidic or basic solutions). The solutions will have varying concentrations and species of dissolved iron and magnesium. Models for colloids will be developed to provide bounding rates for colloidal concentrations and transport characteristics.

The existing dissolution rate model will be updated to incorporate mechanisms for glass. The model will also be modified to include actinide release mechanisms of secondary phases and colloids. Data for these release mechanisms will be provided from the unsaturated drip testing on glass waste forms.

Thermodynamic data will be experimentally obtained to evaluate actinide solubility limits for precipitated actinide species observed in unsaturated tests for a range of temperatures and limited water chemistries.

The waste form testing and modeling work scope supports Milestone M2GH, Waste Form Characteristics Report for SR/LA, presented in Section 7.

3.2.2.9 Waste Package Materials Testing and Modeling

Testing specimens with the same characteristics as the materials comprising the waste package helps in understanding how these materials degrade. The materials to be tested will include high-nickel and titanium-based alloys including crevice corrosion-resistant candidates Alloy 22 (nickel based) and titanium grades 7 and 16 (titanium based with small additions of palladium). Testing will be done in various aqueous solutions and at different temperatures, with samples being removed at planned intervals. Multiple samples will be used for statistical purposes.

Candidate materials will be exposed to potential aqueous environments for at least 5 years. Different configurations of specimens will be used to investigate all forms of corrosion behavior. The test solutions of specimens will be analyzed on a continuing basis for evidence of microbial growth and its effects on corrosion. The results from these tests will be used to develop process models. An enhanced phase stability model will be developed for microbiologically influenced corrosion, pitting and crevice corrosion, stress corrosion, general corrosion and oxidation, and galvanic corrosion.

The degradation of the metallic materials comprising the waste packages will be modeled as a function of time. Modeling will progress from a preliminary level to a more advanced stage as relevant test results become available and can be incorporated.

The specific activities related to the near-field environment address nine areas as listed below:

- Thermogravimetric analysis and controlled humidity experiments
- Long-term corrosion testing of metal barrier candidate materials
- Critical potential measurements, electrochemical galvanic corrosion tests, long-term electrochemical potential tests, and environmentally assisted cracking test
- Long-term tests on microbiologically influenced corrosion
- Phase stability of materials used in waste package designs
- Corrosion testing
- Long-term corrosion testing of basket materials and other criticality control materials
- Long-term corrosion testing of ceramic coated materials
- Engineered barrier system materials testing

Thermogravimetric analysis studies address materials susceptible to accelerated corrosion in humid conditions. The controlled humidity experiments study the degradation of the susceptible materials at less than 100 percent relative humidity and a constant temperature so that degradation of the waste package can be modeled. For the thermogravimetric studies, parameters responsible for inducing the enhanced corrosion will be characterized. The relative humidity at which the transition occurs from dry oxidation to aqueous film electrochemical corrosion will be determined. Determining the critical relative humidity is important to aspects of the repository related to emplacing the waste packages, selecting materials for backfill, thermal loading, and adopting thermal management techniques. The controlled humidity tests complement the short-term thermogravimetric analysis studies.

Long-term corrosion tests are underway for the materials that comprise the outer (corrosion allowance) and inner (corrosion resistant) barriers of the waste package. These materials are being tested under a range of representative repository conditions. Specimens are exposed in the vapor phase (humid conditions), at the water line (partially immersed), and fully immersed (underwater). The interaction between the corrosion-resistant and corrosion-allowance materials (barriers) when the waste package is cracked or breached will also be assessed.

Critical potential measurements involve analyzing conditions that lead to localized corrosion (pitting, crevice corrosion, stress corrosion cracking, and hydrogen embrittlement initiation) on susceptible materials. The critical potentials for passive film breakdown and re-formation (or repassivation) are measured over a wide range of temperature, pH, and electrolyte compositions.

Electrochemical galvanic tests measure the current flowing between materials. These tests provide the basis for determining the effect on the inner barrier when the outer barrier is sacrificially corroded. Key parameters are the compositions of the materials; the steady-state potentials and currents; and the electrolyte composition, temperature, area ratio between the coupled materials, and distance between specimens.

Because the effect of many of these key parameters can be determined in a relatively short time, this activity provides input in determining the parameters that will be used in long-term galvanic corrosion testing.

Tests to determine the materials' susceptibility to stress-corrosion cracking and hydrogen embrittlement use pre-cracked, fracture mechanics type specimens. Results yield quantitative information on the factors that cause a crack to get bigger or grow. This information is used to design the waste package so that conditions conducive to this growth are avoided.

Environmentally assisted cracking tests apply the critical potential concept to stress corrosion cracking and hydrogen embrittlement. Susceptibility to stress-corrosion cracking and hydrogen embrittlement will be evaluated in two ways. The first measures electrochemical potentials using fracture mechanics type specimens. The second applies controlled potentials to smooth tensile specimens continuously pulled in tension at very slow strain rates, while exposed to various corrosive aqueous solutions. The rate of crack growth will be evaluated as a function of the stress intensity factor. These results will supplement the long-term comprehensive testing program.

The potential for corrosion damage to metals from the effects of microbial activities will also be assessed. These studies will have several objectives. The first will be to determine how microbial activities affect the degradation of repository components. The second objective will be to evaluate the resistance of repository materials to microbiologically influenced corrosion. The third and fourth goals deal with assessing the environmental conditions required for microbiologically influenced corrosion, and evaluating the rate of specific and overall microbially mediated corrosion reactions. Corrosion rates will be electrochemically measured. Various combinations of bacterial isolates (solutions containing bacteria) common to Yucca Mountain will be used in these studies. Test specimens will be periodically analyzed to establish the rates of corrosion on metal surfaces.

The long-term stability of corrosion-resistant materials will be tested. The emphasis will be to determine when thermal aging causes phases that can degrade performance. Thermal stability must be understood to develop the waste package design. Thermal aging impacts a material's corrosion behavior, as well as its mechanical response to rockfall and other applied loads. This work characterizes the microstructure of materials and any changes that may occur. Samples of as-fabricated and as-welded container materials from fabrication mockups will be used. The aging studies will be done under appropriate thermal, strain, and environmental conditions to accelerate phase transformations.

The condition of the waste packages after emplacement in the repository will be determined through monitoring and confirmatory testing to determine if they are meeting their expected performance criteria with respect to corrosion. Plans for waste package and confirmatory testing will be developed as part of the LA. One method of monitoring involves using microanalytical devices placed on the package surface. Microanalytical systems, such as specialized sensors, will be developed. For example, sensors to detect local relative humidity at the surface of the waste package will help in monitoring the potential for corrosion during the operational period of the repository.

The current design for the waste package has a thick outer material to protect the inner material from corrosion caused by radiolysis. Estimates have shown that the thickness is sufficient, but more testing will be done to confirm the threshold level for radiolysis-enhanced corrosion. Other subjects for study will be the effect of rockfalls or backfill on corrosion and the condition of localized, through-wall corrosion of the outer barrier that could permit radiolytic processes to speed up the corrosion of the inner barrier. Corrosion rates will be measured as a function of temperature and dose. Further, the corrosion products will be characterized and compared to results from non-irradiated conditions. Long-term tests will be conducted to determine if the corrosion products prevent additional corrosion.

The development of pits in corrosion-resistant materials and the transport of water through these pits will also be studied. The testing will develop pits using standard methods prescribed by the American Society of Testing and Materials. The sizes of the pits and their distribution will be measured.

Another important aspect of this program is studying the degradation over time of the materials comprising the internal structure of the waste package and those controlling criticality. Internal structures will most likely be made of boron stainless steels. Therefore, materials including nickel, chromium, and iron borides and mixed-metal boride will be electrochemically tested. The effect of intense radiolysis on aqueous solutions will be simulated. The effects of temperature, pH, and electrolyte composition will also be determined, as will the expected relationships between the microstructure and electrochemical behavior.

Finally, models will be developed to predict the long-term performance of materials that contain a neutron-absorbing material. The model will address three issues: the nature of corrosive attack on the boron stainless steel, how the attack is partitioned between phases, and if the properties for controlling criticality can remain effective for long periods.

The viability of ceramic materials as an alternative or augmentation for the inner or outer barriers of the waste package or for use in drip shield or borehole lining applications may be evaluated. Ceramics would best be used as coatings on metallic material. Tests will be conducted to determine the adhesive strength of the coating and its ability to withstand thermal and handling loads. Critical factors in assessing the effectiveness of ceramic coatings include the ability to maintain an impervious barrier of protection for the material being coated (substrate). The permeability and density of the coatings as a function of the type of ceramic and the way it is applied will also be determined. Samples of various thickness, structure, and composition will be exposed to corrosive conditions to find out how much protection the coatings provide. Of particular importance is how porous or cracked regions of the coating affect the long-term behavior of the substrate metal.

Testing will be performed to predict the behavior of various materials being considered for the repository engineered barrier system. The model for how groundwater will interact with corrosion products and concretes will be developed with an emphasis on biological impacts from microbes that exist or could be brought into the system. Optimization modeling may be used to determine aspects of microbial activity. These results can be incorporated into geochemical models to develop a repository model. Test results will be incorporated into the model. This work supports subsurface and waste package design and performance assessment analyses of containment and radionuclide release. This work also complements the work on the characteristics of the effects of man-made materials on chemical and mineralogical changes in post-emplacement environments, and supports the design features and alternatives evaluations by evaluating the effects of committed underground materials.

The waste package material testing and modeling work scope supports Milestone M2GY, Engineered Materials Characteristics Report for SR/LA, presented in Section 7.

3.2.3 Surface

This section describes the planned technical work on repository surface design to be completed before the LA is submitted. Because surface facilities deal almost exclusively with the preclosure safety case, this section differs from the previous two sections describing planned work on the repository subsurface and waste package design. The systems and facilities in the surface portion of the repository are based in great part on proven, existing systems and facilities, either in the commercial reactor or DOE environment. As such, there is confidence in the design to safely receive and package nuclear waste for emplacement underground. Engineering efforts have been focused on applying the industry proven systems and these efforts will transition to developing of unprecedented nuclear safety related systems specifically tailored to the repository. This effort will be based on the alternative design that will be selected before the LA is submitted.

The current design will be modified to incorporate selected features and achieve the level of design adequate for the LA. A list of design documentation expected to be produced to support the LA will be generated in response to general DOE guidance. If necessary, operations, methods, and functions of material handling and process equipment may be evaluated through the design process. This evaluation may include producing and revising drawings, reports, calculations, and analyses. The level of detail in each of these products will depend primarily on the importance to supporting the preclosure safety strategy (i.e., protecting the public, workers, and the environment from radiological hazards) and the extent to which the design is similar to currently licensed or proven designs.

3.2.3.1 Introduction

Within this section, a general description of surface facilities and their functions includes some discussion of design support for the preclosure safety case. Where applicable, the discussion is tied to the NRC key technical issues and described in Section 4.3.3. An overview of the design alternatives being evaluated and some of the potential effects on surface facility design precedes a discussion of the types of design documentation and engineering products that are expected to be required to contribute to the evaluating of the alternatives and advancing the VA design to a level adequate to support the LA. The actual engineering design work sections are organized by primary surface facilities:

- Carrier Preparation Building
- Waste Handling Building
- Waste Treatment Building
- Sitewide systems and support

Table 3-8 shows the relationships between the work to design these facilities, the preclosure and postclosure safety cases, and the references to cost and schedule information. The discussion also covers work to support additional program and licensing activities. Work activities within each section support DOE milestones presented in Section 7.

3.2.3.2 Surface Facilities

The primary functions of the surface facilities will be to receive shipments of waste, prepare and package the waste for emplacement underground, and transfer the packaged waste to the underground transporter. The majority of the related activities occur in structures built above ground in the repository North Portal area. The surface area where radioactive material will be handled is called the radiologically controlled area; the area outside of the controlled area that supports general site operations is referred to as the balance of plant area. A detailed description of surface operations and the design for the VA is included in Volume 2, Sections 4.1 and 6.2. These sections also contain a number of figures that depict the location and operation of the surface facilities.

A major design goal for the repository is radiological protection of personnel and the environment, both during waste handling operations and after the waste has been emplaced underground. The level of radioactive protection that drives many of the design requirements is prescribed by NRC regulations. The preclosure safety case discussed in Section 2.3 identifies the systems, structures, and components important to radiological safety that are needed to comply with the regulatory requirements and the offsite exposure limits prescribed in 10 CFR 60. Some design of the surface facilities and associated systems and equipment is expected to be needed to define critical elements that support the preclosure safety case, as well as to support other technical work that is required by 10 CFR 60 but does not directly support either preclosure or postclosure. Design work for surface structures, systems, and components will continue to evolve and support the final stage of the site-characterization phase, while supporting the change in focus from repository scientific investigation to design, engineering, and performance assessment. While this work does not specifically address the NRC key technical issues identified in Section 4.3.3, the work will provide the design solutions necessary to support the structures, systems, and components to be described in the LA. This work will coincide with the future technical work and statutory prelicensing steps, and support activities described in

Table 3-8. Application of Planned Surface Design Work to the Postclosure and Preclosure Safety Cases

Work Category (from text)	Postclosure Safety Case[1]	Elements of the Postclosure Safety Case[1]						Preclosure Safety Case[3]	PSS Activities[4]
		1. Expected Postclosure Performance		2. Design Margin/ Defense in Depth	3. Disruptive Proc./ Events	4. Natural & Man-Made Analogs	5. Performance Confirmation		
		Principal Factors Supported by Technical Work	Priority to SR/LA[2]						
Surface Facility (3.2.3.2)		N/A						√	2024 3040 3070 6103
Design Documentation and Engineering Products (3.2.3.3)		N/A						√	2024 3040 3070 6103
Carrier Preparation Building Design (3.2.3.4)		N/A						√	2310 2392 2393 2403
Waste Handling Building Design (3.2.3.5)	√	Integrity of outer carbon steel waste package barrier Integrity of inner corrosion-resistant waste package barrier Integrity of spent nuclear fuel cladding	1 3 2					√	2310 2392 2393 2403
Waste Treatment Building Design (3.2.3.6)		N/A						√	2310 2392 2393 2403
Sitewide Systems and Support Design (3.2.3.7)		N/A						√	2310 2392 2393 2403
Support Activities (3.2.3.8)		N/A						√	2019 2023 2403 3040 3070 6103

[1] Postclosure safety case described in Section 2.2.
[2] Postclosure work prioritization for site recommendation and LA described in Section 2.2.4.
[3] Preclosure safety case described in Section 2.3.
[4] Project Summary Schedule (PSS) activities are the basis for LA plan cost and schedule, in Sections 6 and 7.

Section 1 and included in the list of program milestones in Section 7.

In conjunction with design in the four primary surface facility areas, near-term tasks associated with design features and design alternatives will be completed to support selecting the initial LA design. The design areas will factor in and support these evaluations as a normal part of the design process, working in concert to advance the current reference design while supporting the quantitative evaluation of alternatives described by Volume 2, Section 8. Specific alternative categories requiring surface facility considerations are as follows:

Category 1. Containment Within the Engineered Barrier System

- Drip shields or waste package ceramic coatings–options for application of waste package or drip shield ceramics in the surface facility; preparation for subsurface transport and emplacement

- Backfill/Richards' barrier–surface handling, staging, inspection, or storage of materials

Category 3. Integrated Effects of Thermal Loading

- Thermal Management
  - Pre-emplacement aging; lag storage options
  - Fuel blending
  - Rod consolidation options

Category 4. Waste Package Production and Emplacement Operations

- Waste Handling Building waste package production line capacity and throughput
- Facilities for waste package fabrication and closure options

The focus of surface facility considerations emphasizes feasibility of operational concepts, radiological safety implications, cost implications, and other information pertinent to the selected evaluation criteria. The combination of features, when evaluated for a given alternative, could potentially impact the facility layout, site layout, throughput capacity, and material handling processes and equipment.

Additionally, several potential program level enhancements that influence performance of the design alternative evaluations may be assessed, as input to either the alternative evaluation criteria, or the design alternative definitions. These enhancements include items such as modular design of the facilities to spread the construction costs. The results of these evaluations will also support selecting the initial LA design, followed by more detailed design analyses necessary to support the LA safety arguments.

The work on the evaluation of alternative designs and options will support Milestones M2MP and M2MR under licensing and M2MQ, M2MT, and M2MU under design in the milestone table in Section 7.

3.2.3.3 Design Documentation and Engineering Products

The design described in Volume 2 is supported by analyses, reports, and other referenced documentation. As required, operations, methods, and functions of material handling and process equipment will continue to be evaluated through the design process. This evaluation will include producing and revising of any necessary diagrams, drawings, reports, calculations, and analyses. Because the level of detail in each of these products will depend greatly on the importance to supporting the preclosure safety strategy (i.e., protecting the public, workers, and the environment from radiological hazards) and the extent to which the design is similar to currently licensed or proven designs, the documentation for various systems, structures, and components will follow the design prioritization, or binning, process described in Section 2.3. In general, any documents that may be produced dealing with support facilities, site utilities, and conventional support areas of nuclear-related facilities will require the lowest level of detail and will be Bin 1. Documents dealing with operational and primary support areas of nuclear-related facilities, and selected site-support systems having precedent, will be Bin 2. Documents dealing with selected waste handling equipment, and systems without precedent, will require extensive detail and will be Bin 3.

A database will relate the work of each engineering discipline (architectural; instrumentation and control; civil; electrical; fire protection; heating, ventilating, and air conditioning; mechanical; nuclear; piping; process; safety and security; and structural) to the level of detail needed in each required design document, based on general DOE guidance, the bin number of the specific activity, and the type of document required.

The actual documents to be produced and the level of detail will be determined as the design progresses. Documents for Safety Class 1 or 2 structures, systems, and components are expected to have the following general content:

- The documents will range from depicting basic site layout and schematic building plans to enlarged layouts of rooms, systems, processes, and equipment in those areas of design related to radiological safety.
- Designs in areas related to radiological safety must be completed to a sufficient level of detail to enable NRC to determine that the repository, as designed, would adequately protect public and worker health and safety.
- The current designs will be reviewed and developed to provide the level necessary to adequately address design margin and defense in depth to include greater margins of safety than necessary.
- Disruptive processes and events such as earthquakes, drop accidents, and human intrusion will also be considered.

Design documentation supporting the evaluation of disruptive events in the TSPA will incorporate the NRC key technical issue of Structural Deformation and Seismicity (Section 4.3.3.2) and its related subissues. Note that consideration of disruptive events is a subissue associated with the key technical issue concerning activities associated with the Development of the EPA Standard for Yucca Mountain (Section 4.3.3.8).

The following sections summarize the engineering work in the four primary surface facility areas that will be completed to support site recommendation, alternative selection, waste form characteristics, engineered materials characteristics, and submittal of the LA. This work will support Milestones M2NN, M2MR, M2KC, and M2NV under licensing; and Milestones M2HW, M2GX, M2MQ, M2MT, M2MU, M2MX, M2GH, M2GY, M2MV, MTMX, M2GH, M2GY, and M2MW under design in the milestone table in Section 7.

3.2.3.4 Carrier Preparation Building Design

Incoming shipping casks containing the waste will arrive on rail or truck carriers. In the Carrier Preparation Building, the casks will be inspected and cleaned, and the carrier's external personnel barriers and tie-downs will be removed to expose the cask and allow the collision impact limiters at either end of the cask to be removed or extracted.

The Carrier Preparation Building is based on proven technology and design with precedent at existing facilities; therefore, no additional design work is planned prior to LA.

3.2.3.5 Waste Handling Building Design

After the shipping casks go through the Carrier Preparation Building, they will be opened in the Waste Handling Building and the waste forms will be placed in containers that are designed to be emplaced in the subsurface portion of the repository. The Waste Handling Building will be the largest surface structure, and will house the majority of the waste handling system, as well as a number of support systems and subsystems described in Volume 2. Work on the Waste Handling Building design is expected to include the following:

- Development of the design of the systems and equipment within these facilities must accommodate various shapes and sizes of waste, some of which weigh many tons. Actual waste receipt information, when received, will impact this design element.
- Analyses of the systems, structures, and components in the primary material handling systems, which will handle the spent nuclear fuel and high-level radioactive waste coming into the repository.

Descriptions of these mechanical systems and figures depicting the path that selected canisters containing waste forms takes within the systems are included in Section 4.1 of Volume 2. Some canisters containing waste forms will be placed directly into disposal containers for emplacement, while others will be opened under water and their contents removed for insertion into disposal containers. The systems will be designed to provide for transferring waste forms safely under either condition.

Most of the waste handling systems are expected to require Bin 2 design documentation, although a few unprecedented systems and components are expected to be Bin 3. The design of the Waste Handling Building structure is expected to be primarily Bin 2.

Because the surface facilities will be decommissioned and removed after the repository is closed, they do not directly affect the postclosure case. However, operations completed in the surface facilities during handling and preparation of the waste must meet the criteria for long-term disposal in the repository. Therefore, functions of the waste handling systems will indirectly support the postclosure safety case.

3.2.3.6 Waste Treatment Building Design

Any solid or liquid low-level radioactive waste that has been generated during waste preparation and handling operations will be processed by the site generated radiological waste system. The majority of this system will be located in the Waste Treatment Building.

The Waste Treatment Building is based on proven technology and design with precedent at existing facilities; therefore, no additional design work is planned prior to LA.

3.2.3.7 Sitewide Systems and Support Design

For the VA reference design, emphasis was placed on designing the facilities, systems, and equipment that will handle the waste casks/carriers, prepare the shipping casks for unloading the waste, load the waste into the disposal containers, and transport the containers to the subsurface. A description of this work is included in Volume 2, Section 4.1.

The Transporter Maintenance Building is a site-support facility located in the controlled area. The majority of the remaining site support facilities and systems are in the balance of plant area. These include the following:

- Administration building
- Medical center
- Central warehouse
- Central shops
- Mockup building
- Utility building
- Visitors building

Each of the three main surface structures (the Carrier Preparation Building, the Waste Handling Building, and the Waste Treatment Building) incorporates some or all of the following support systems:

- Ventilation
- Cooling water distribution
- Potable water distribution
- Electrical power distribution
- Uninterruptible power distribution
- Bottled gas supply
- Breathing air system
- Fire suppression system
- Utility, facility, and environmental monitoring
- Radiological monitoring
- Telephone, public address, and teleconferencing
- Sanitary sewer
- Lighting
- Emergency alarm

- Pool water system
- Decontamination systems
- Material control and accountability

The basic functions of each of these systems are based on proven technology and design with precedent at existing facilities; therefore, no additional design work is planned prior to LA.

3.2.3.8 Support Activities

Section 5.2 discusses activities other than field construction and operations necessary to support repository development: information technology, program information management, systems engineering, quality assurance, project management, institutional interactions, administrative services, and training. Integration of each of these activities will continue during repository design.

For example, as the repository design and alternative selection evolves to a level adequate for the LA, the systems engineering approach described in Section 3.2 will be used to produce a list of structures, systems, and components necessary to generate system design documents. These documents will include the design criteria required for completing the preclosure radiological safety case for surface facilities. Criteria backup sheets will also be prepared to support these system design documents and to provide design input to systematically identifying design basis events to determine the safety classification of the structures, systems, and components. These work activities will support Milestone M2HW under design in the milestone table in Section 7.

In addition to design documents, a number of other products, such as design basis event analyses and quality assurance classification analyses, will be developed. Input, review, and production support for the EIS will also be provided. These and similar activities will support Milestone M2MW under design and Milestone M1AX under EIS and environmental compliance in the milestone table in Section 7.

3.3 PERFORMANCE ASSESSMENT

Performance assessment work has been planned to support completion of the work identified in Section 2. The purpose of this work is to update the TSPA completed for the VA for use in the site recommendation and the LA. The TSPA-VA analyses represent a significant improvement over previous TSPA iterations. For each of the principal factors associated with the TSPA model components, the potential significance of some aspects of the model uncertainty was not analyzed in the TSPA-VA (Volume 3, Section 3.6.5.1). The planned work for site recommendation and the LA will focus on reducing the remaining uncertainties for these principal factors affecting the long-term performance of the repository. Models used in TSPA-VA will be refined to incorporate a relatively small set of new information. Planned work will also evaluate approaches for achieving defense in depth and margin. This work will support the evaluation of alternative designs, design features, and design options. Finally, the planned work addresses key technical issues identified by NRC.

The planned work is organized into the following categories:

- Model abstractions
- TSPA analyses
- Design support

3.3.1 Model Abstractions

Because of the overall uncertainty and complexity of the repository system, the process models used in TSPA analyses must be simplified to reproduce and bound the underlying detailed process model. This simplification or abstraction retains the basic form of the process model. These model abstractions are required to maximize the use of finite computational resources while maintaining a sufficient range of sensitivity and uncertainty analyses. Once the abstracted models are developed, they are used for a performance assessment of the repository system.

The site investigation and design work carried out after the model abstraction done for TSPA-VA will result in process models that are revised to a greater or lesser extent in light of the new information that becomes available. These improved process models will be abstracted into improved model components to be used in the TSPA for the site recommendation and the LA.

Work planned to complete the abstraction process is organized into the following six categories:

- Unsaturated zone flow and transport
- Near-field environment
- Waste package
- Waste form and engineered barrier system
- Saturated zone flow and transport and biosphere
- Disruptive events

These components are then combined to assess the expected behavior of the system. The combined components include alternative conceptual models, variability and uncertainty in parameters and processes, and a sufficient level of detail to represent the aspects of each process important to performance. This work is described in Section 3.3.2.

The first step in the process of model abstraction is to identify uncertainties in the TSPA-VA and each supporting model. This effort will consider comments from the expert elicitations completed for the VA, comments from the TSPA-VA peer review panel, and comments from organizations such as NRC and the Nuclear Waste Technical Review Board. Model enhancements will be defined along with approaches to addressing the identified uncertainties and approaches to treating defense in depth and margin for each of the models.

The results of this work will support Milestone M2JH, TSPA-SR/LA Methodology and Assumptions Document.

All of this work addresses the NRC key technical issue on the TSPA and Technical Integration. Specifically, the model abstractions will provide part of the technical basis to address the following subissue of this key technical issue: Are the major components of DOE TSPA methodology (i.e., model abstractions; probability and consequences of relevant features, events, and processes; parameter and model uncertainties; and bounding assumptions) sufficiently comprehensive to provide a defensible safety case?

Unsaturated Zone Flow and Transport. The flow of water and transport of radionuclides in the unsaturated zone at Yucca Mountain have been shown to play an important role in how the potential repository will perform. Water seeping into the drift and onto the waste packages can initiate and accelerate their degradation. Fractures in the host rock provide rapid pathways from the repository to the water table that can decrease the transport time of radionuclides to the water table. The primary processes affecting radionuclide transport through the unsaturated zone are radionuclide decay, advection of the bulk fluid, diffusion within the liquid phase, and partitioning to other phases. Assessing these events relies on process models of flow and transport in the unsaturated zone that have been tested or calibrated against available data at Yucca Mountain. The models of flow in the unsaturated zone are separated into two categories according to scale; far-field models of groundwater flow and drift-scale models for seepage. These models are then abstracted into unsaturated zone flow and seepage models for the TSPA.

This activity refines the abstractions of process models for climate, infiltration, percolation, ambient seepage, and transport through the unsaturated zone. The primary focus will be placed on transport through the unsaturated zone. New information from studies of unsaturated zone flow and transport processes and updated process models will be evaluated. This activity will develop and document the technical basis for the refined abstractions that will be used in the TSPA for the site recommendation and the LA.

Model enhancements that have the potential for improving the confidence in future assessments of repository performance include improving the technical basis for the timing and duration of climate states and for the abruptness of transitions from one climate to another. The basis for modeling of infiltration could be expanded to include parameters such as temperature. To address the uncertainty in net infiltration, a probabilistic model could be run to better quantify the uncertainties in net infiltration. Estimates of future infiltration could be improved by explicitly including variables, such as temperature and vegetation. Infiltration data from natural analogs sites, such as Rainier Mesa, could be used to increase the confidence in the infiltration model.

Improved integration of geochemical, isotopic, and temperature data could be used to evaluate the importance of localized flow channeling. The current model for perched water assumes that the water is perched on a very low-permeability layer and that flow is forced to go around this layer. Alternative models, such as matrix flow out of the bottom of the low-permeability layer, could be considered.

Field data from analog sites, such as Rainier Mesa or Apache Leap, could be used to calibrate the calculated values of the seepage fraction. Other areas of potential improvement include the effects of localized flow channeling on seepage, the stability of seep locations over time, the effect of drift collapse on seepage, and the effect of episodic percolation.

The results of field and laboratory tests will be used to improve models of unsaturated zone transport. Potential enhancements include consideration of the effects of thermal, hydrologic, and chemical alteration; colloid filtration; sorption in fractures; and matrix diffusion.

At the start of this activity (2220), a workshop will be held to review new data and refined process models from the testing program and comments and recommendations from external reviews of the TSPA-VA. A complete list of technical issues, including uncertainties in the relevant principal factors, will be developed and prioritized and a plan will be completed to define the specific analyses for the model abstractions.

This work contributes to addressing NRC key technical issues on Unsaturated and Saturated Flow Under Isothermal Conditions, and Radionuclide Transport (Sections 4.3.3.9 and 4.3.3.10) by incorporating new data and the results of refined process modeling into the abstractions. This work also addresses the first element of the postclosure safety case (assessment of expected postclosure performance) by reducing uncertainties in the evaluation of flow and transport through the unsaturated zone. Some of the work also supports the fourth element of the postclosure safety case by incorporating field data from natural analog sites.

The work supports Milestones M2JJ, TSPA-SR/LA Base Case Results Document, and M2JG, Complete TSPA-SR/LA Documentation.

Near-Field Environment. The near-field environment includes ambient geochemistry and analyses of the thermal effects on the ambient geochemistry, hydrology, and mechanical properties of the rocks surrounding the drifts. The model for the near-field geochemical environment affects, and is affected by, all other processes within the engineered barrier system, including the coupling to thermal-hydrologic processes. This model focuses on major element geochemistry within the emplacement drifts. Key input to the model includes the major-element composition, and thermodynamic and kinetic coefficients of introduced in-drift materials; the thermal-hydrologic behavior of the repository; and the gas-phase and aqueous-phase compositions entering the drift through time. These parameters depend on the rate of percolation (flux), the reaction and reflux of condensate, seepage into drifts, and the design of the repository. All these parameters will influence the ability to discern and bound the actual near-field geochemical environment when using models.

The repository horizon in the unsaturated zone at Yucca Mountain contains in situ pore liquid water and gas that will react to the heat from radioactive wastes placed in the repository. The thermodynamic environment in the emplacement drift depends strongly on the decay characteristics from each waste package. Quantification of the thermodynamic environment within the drifts describes the conditions in the drift and surrounding the waste packages. This description includes temperature, relative humidity, air mass fraction, and fluid flow time histories near the waste packages and into the drift. The conditions in the emplacement drift affect how the repository performs by influencing the overall lifetime of the waste packages.

This activity refines the abstraction of process models for thermal hydrology, ambient geochemistry, and near-field environment. Results from field thermal tests and laboratory tests and updated process models will be evaluated. Process modeling will have assessed design options and features, such as drip shields, backfill, and Richards' barriers to evaluate their contribution to performance (Section 2.2). The effects of heating and excavation on flow, dryout by ventilation, and water diversion by line loading will also be evaluated. The results of these abstractions will also address seepage into drifts, dripping onto waste packages, humidity and temperature at the waste package, and chemistry on the waste package. The results of this work will be documented to provide the technical basis for processes in the near-field environment for the TSPA for the site recommendation and the LA.

Potential modeling enhancements include improvements to the conceptual model of flow and the treatment of coupled processes, and improved coupling of thermal-hydrologic effects with the mountain-scale unsaturated zone flow and transport models. Additional analyses could include analyses of the effects of infiltration on temperature. Results from the single heater and the drift-scale tests will provide an important source of data to refine the thermal-hydrologic models. These data will provide insights into the effective hydrologic properties during various stages of heating and on the spatial and temporal extent of mechanical and chemical changes in the fracture flow system surrounding the heated drift.

Improvements to the analysis of the effects of concrete-modified water could be implemented to help constrain the uncertainties in the current models of the near-field geochemical environment. Improvements could include better definition of thermal-chemical data for phases in the cement system, changes in gas composition in the unsaturated, thermally perturbed system, and development of a two-phase, reactive transport model of concrete alteration. Alternative models for water in the drift may be needed to evaluate precipitate and/or salt build up on the waste package. Coupled models would benefit from the incorporation of more data on drift materials, including their physical-mechanical evolution and chemical changes to the hydrologic properties of the engineered materials. The updated near-field geochemical model could consider the heterogeneity of water interacting with the drift environment, CO_2 evolution from water and minerals coupled to gas flow, thermal aging of emplaced materials, and microbial activity.

At the start of this activity (2235), a workshop will be held to review new data and refined process models from the testing program and comments and recommendations from external reviews of the TSPA-VA. A complete list of technical issues, including uncertainties in the relevant principal factors, will be developed and prioritized and a plan will be completed to define the specific analyses for the model abstractions.

The work contributes to addressing NRC key technical issues on Thermal Effects on Flow and Evolution of the Near-Field Environment (Sections 4.3.3.3 and 4.3.3.5) by evaluating the heat-driven redistribution of moisture through a partially saturated porous media, by evaluating the seepage of water into the emplacement drifts, and by reducing uncertainties in the near-field geochemical environment. This work also addresses the first element in the postclosure safety case (assessment of expected postclosure performance).

The work supports Milestones M2JJ, TSPA-SR/LA Base Case Results Document, and M2JG, Complete TSPA-SR/LA Documentation.

Waste Package. The waste package is one of the most important components of the potential repository system. Because the waste package will eventually degrade, water in nearby rock will move into the waste package, dissolve radionuclides, and carry them away from the package. The degradation of the waste package under relevant repository conditions is a key factor in evaluating releases of radionuclides from the repository. To assess the rate at which waste packages fail, the model for waste package degradation must include information on the designs of both the repository and the waste package, the near-field environment surrounding the waste packages, and the degradation characteristics of the materials comprising the waste packages.

Currently, the waste package designs all have an inner layer of highly corrosion-resistant material. Therefore, mechanisms for localized corrosion, such as pitting, are a key performance issue. Localized corrosion is believed to be the most likely barrier failure mode that could significantly impact the performance of the engineered barrier system. Fundamental localized corrosion mechanisms and abstracted representations used in performance modeling include significant uncertainty and alternative conceptual models. Improving these models will provide a better, and more accurate, understanding of how the overall system performs. Abstracted representations of localized corrosion will be improved based on the results of corrosion testing and the development of mechanistic models. These improved abstracted representations will then be incorporated into performance assessment models for the engineered barrier system.

This activity refines the abstraction of the process model for waste package degradation. New data from the waste package materials testing program, results of material selection for the initial waste package for the site recommendation and the LA, and the evaluation of design features will be used to update the abstraction of waste package degradation processes. The work will be documented as part of the technical basis for TSPA for the site recommendation and the LA.

Assuming no change to the reference design, improved analysis of the long-term corrosion of carbon steel under dripping conditions and in the presence of salt deposits would reduce the uncertainty in the long-term corrosion of carbon steel.

Similarly, improved understanding of the local chemical and electrochemical conditions on the inner barrier and additional data on the behavior of Alloy 22 would reduce the uncertainty in the corrosion model for Alloy 22. Reduction in the uncertainties in the general corrosion rate, the pitting and crevice corrosion rates, the threshold for initiation of pitting and crevice corrosion, and an improved understanding of the stifling process for pitting and crevice corrosion would improve the overall understanding of corrosion for Alloy 22. Others topics that would benefit from reduction in uncertainty include potential galvanic coupling, incomplete annealing of welds, microbiologically induced corrosion, and the long-term structural integrity of the waste package.

The amount of water that actually enters the waste package once it is breached is highly uncertain. Reducing this uncertainty would improve the TSPA analyses because the amount of water contacting the waste form is a significant factor in releases from the engineered barrier system and is conservatively analyzed in the TSPA-VA. Analysis of alternative models of water flow in the degraded waste package would refine the evaluation of the mode in which water contacts the waste form.

At the start of this activity (2195), a workshop will be held to review new data and refined process models from the testing program and comments and recommendations from external reviews of the TSPA-VA. A complete list of technical issues, including uncertainties in the relevant principal factors, will be developed and prioritized and a plan will be completed to define the specific analyses for the model abstractions.

The importance of the assessment of waste package degradation is reflected in the NRC key technical issue on Container Life and Source Term (Section 4.3.3.4). This work also addresses the first element of the postclosure safety case by reducing uncertainties in the overall performance of the inner and outer waste package barriers.

The work supports Milestones M2JJ, TSPA-SR/LA Base Case Results Document, and M2JG, Complete TSPA-SR/LA Documentation.

Waste Form and Engineered Barrier System. If the waste package fails, the waste form inside is potentially exposed to air, water vapor, and flowing water. This scenario does not imply that all of the radionuclides are available for transport out of the engineered barrier system. First, because only a small percentage of the total radionuclide inventory is mobile enough and has a long enough half life to be transported out of the waste package if it is breached. Second, because the waste forms themselves are barriers to the release of radionuclides. For radionuclides to be released, the waste forms would also have to degrade and water would have to be available to carry the radionuclides through the engineered barrier system into the unsaturated zone in the host rock.

This activity refines the abstraction of process models for cladding degradation, waste form degradation, colloid formation and stability, and solubility and transport through the engineered barrier system to reduce uncertainties in the rate of degradation of the waste forms and the mobilization of radionuclides. The oxidation and dissolution of spent nuclear fuel is predicted based on semi-empirical conservative models. The models will be based on the latest experimental data available from ongoing experimental programs and from the international community. A model that accounts for changes in the waste form, such as the intrinsic dissolution rate of spent nuclear fuel as a function of a state of oxidation and available surface area for oxidation, will be updated in conjunction with the testing program. The work will focus on the evaluation and abstraction of the updated process models for cladding degradation, colloid formation, and studies of solubility. The work will be documented as part of the technical basis of the TSPA for the site recommendation and the LA.

Laboratory testing will contribute to refined modeling of the dissolution processes. Tests on spent nuclear fuel will evaluate the effect of dripping rate of water, the integrated flow volume, the estimated surface area of the sample, the estimated time the water stays in contact with the surface of the spent nuclear fuel, and the chemical and radionuclide states in aqueous solutions. Testing of the glass waste form will be conducted to attempt to simulate long-term behavior over short intervals by testing at elevated temperatures and at high surface-area-to-volume ratios. These data will provide input to refined dissolution models.

Explicit analysis of diffusive transport out of a locally failed fuel rod would improve the understanding of the transport resistance of failed Zircaloy cladding. Reduction in the uncertainty of cladding failure mechanisms would improve the current understanding of cladding degradation.

Laboratory tests of speciation, solubility, and sorption will be completed to refine analyses of key radionuclides. Laboratory tests and field tests at Busted Butte will be used to update the conceptual understanding and modeling of the formation and stability of colloids and their interactions with radionuclides.

Evaluation of natural analogs (such as the natural reactors at Oklo in Gabon, Africa) may help to reduce uncertainties in transport processes. Analysis of mechanism for sorption in the engineered barrier system would improve the modeling of sorption processes in the engineered barrier system.

At the start of this activity (2190), a workshop will be held to review new data and refined process models from the testing program and comments and recommendations from external reviews of the TSPA-VA. A complete list of technical issues, including uncertainties in the relevant principal factors, will be developed and prioritized and a plan will be completed to define the specific analyses for the model abstractions.

This work contributes to addressing the NRC key technical issue on Container Life and Source Term (Section 4.3.3.4). This key technical issue focuses on the lifetime of the waste package, the rate at which radionuclides are released from breached waste packages, and the transport of radionuclides out of the engineered barrier system. This work also addresses the first element in the postclosure safety case by reducing uncertainties in the degradation of the waste form and transport through and out of the engineered barrier system. Some of the work will also support the fourth element of the postclosure safety case by incorporating data from natural analogs.

The work supports Milestones M2JJ, TSPA-SR/LA Base Case Results Document, and M2JG, Complete TSPA-SR/LA Documentation.

Saturated Zone Flow and Transport and Biosphere. Radionuclides escaping the repository, transported in dissolved or colloidal form, are expected to travel downward through the unsaturated zone to the saturated zone where they could be a source of contamination for the biosphere. Important processes to be considered in saturated zone flow and transport include advective transport, diffusion, dispersion, and geochemical retardation. The saturated zone flow and transport component of the analysis evaluates the migration of radionuclides within the saturated zone, from their introduction at the water table near the repository to the point of release to the biosphere. The saturated zone flow and transport analysis is linked to the biosphere analysis via the simulated time history of radionuclide concentration in groundwater produced from a hypothetical well located 20 km (12 miles) from the repository.

The final step in the analysis is modeling the movement of radionuclides from the geosphere into the biosphere, where flora and fauna are present. If radioactive isotopes are introduced into the biosphere, they are exposed to various physical, chemical, and biological processes, some of which can result in radiation exposures to humans. It is important to know how radionuclides move through the biosphere and how they might eventually impact human inhabitants of the region. The primary measure of the repository's performance is the annual dose of radiation that would be received by an individual living in the region.

This activity refines the abstraction of process models for transport through the saturated zone and the biosphere. New site data on hydraulic parameters from testing at the C-well complex, on saturated zone hydrochemistry, and results of testing in wells in the Amargosa Valley and Nevada Test Site and updated process models for saturated zone flow and transport will be evaluated. Additional environmental data and the updated biosphere process model will also be evaluated. This work will provide the technical basis for the updated abstraction of saturated zone flow and transport and biosphere for TSPA for the site recommendation and the LA.

New data will be available to refine the model abstractions. Additional potentiometric and hydrochemical data from approximately 10 km (6 miles) to 20 km (12 miles) downgradient from the repository will be available to help constrain flow paths in this part of the saturated model. Additional geochemical and isotopic data from the saturated zone will help constrain conceptual models of groundwater flow. Inferences from analog sites and possibly from natural solute tracers in the saturated zone at the site may help to reduce the uncertainty in the dilution factor and the vertical transverse dispersivity. The effects of changes in climate on the saturated zone could be improved by considering additional discharge locations and the effect of additional recharge at Yucca Mountain, Forty Mile Wash, and regions upgradient from Yucca Mountain. Modeling of colloid-facilitated transport in the saturated zone needs to be improved.

An improved definition of the interface between the geosphere and the biosphere, taking into account natural discharge points and dilution from mixing contaminated water with uncontaminated water during well pumping would strengthen the calculations of dilution. Modeling would benefit from an improved definition of the critical group and individual receptor. Analysis of the influence of climate change on the biosphere could be considered. Other factors that could improve the modeling of biosphere transport and uptake include assessments of the long-term buildup of radionuclides at natural discharge locations, well withdrawal effects, assumptions of well locations, site-specific data on drinking water ingestion, leafy vegetable ingestion, and meat ingestion, and evaluation of radionuclide build up in the soil.

At the start of this activity (2176), a workshop will be held to review new data and refined process models from the testing program and comments and recommendations from external reviews of the TSPA-VA. A complete list of technical issues, including uncertainties in the relevant principal factors, will be developed and prioritized and a plan will be completed to define the specific analyses for the model abstractions.

This work contributes to addressing NRC key technical issues on Unsaturated and Saturated Zone Flow Under Isothermal Conditions and Radionuclide Transport by providing refined model abstractions (Sections 4.3.3.9 and 4.3.3.10). This work also addresses the first element of the postclosure safety case (assessment of expected postclosure performance) by reducing uncertainties in the evaluation of flow and transport through the saturated zone and the biosphere.

The work supports Milestones M2JJ, TSPA-SR/LA Base Case Results Document, and M2JG, Complete TSPA-SR/LA Documentation.

Disruptive Events. Previous work has identified four disruptive events that might affect the long-term performance of the repository: volcanic activity, earthquakes, inadvertent human intrusion, and nuclear criticality.

Studies of volcanoes and earthquakes have reached a level of maturity for which only limited new information is required. The remaining work will focus on completing any refinements to the hazard analyses that are required to provide a documented technical basis of the effect of these disruptive events on the performance of the repository system.

Based on guidance from EPA and the National Academy of Science, models will be developed to evaluate the consequences of inadvertent human intrusion at the Yucca Mountain site. These models will evaluate direct effects such as surface releases caused by drilling into waste packages and indirect effects from drilling rapidly transporting radionuclides into the saturated zone. The performance assessment will use scenarios that have the potential for either high consequences or high relative probabilities of occurrence and develop model parameter distributions that reflect the variabilities and uncertainties in the conceptual models and data. This work will provide the technical basis for the assessment of inadvertent human intrusion.

Nuclear criticality could occur if the engineered control measures in the waste package fail or if fissile material in the waste forms a critical configuration in the surrounding rock. Work is required to refine the consequences of potential criticalities and to document the technical basis for the assessment of this potentially disruptive event.

At the start of this activity (2175), a workshop will be held to review new data and refined process models from the testing program and comments and recommendations from external reviews of the TSPA-VA. A complete list of technical issues will be developed and prioritized and a plan will be completed to define the specific analyses for the model abstractions.

This activity addresses the NRC key technical issues for Igneous Activity and Structural Deformation and Seismicity by documenting the hazard associated with these disruptive events. This work also addresses the third element of the postclosure safety case (disruptive processes and events) by providing the documentation of the extent of the hazard that potentially disruptive processes and events pose at the Yucca Mountain site.

This work supports the completion of Milestone M2JG, Complete TSPA-SR/LA Documentation.

3.3.2 Total System Performance Assessment Analyses

Once the model abstractions have been refined, they must be appropriately linked for the assessment of system performance. The TSPA analyses will then be completed to evaluate the expected behavior of the repository system. A wide range of sensitivity and uncertainty analyses will also be conducted to evaluate the significance of the uncertainties in parameters, conceptual models and the 19 principal factors. The sensitivity and uncertainty analyses will help to identify the key site and design features of the repository system that have the greatest impact on repository performance.

Before the TSPA analyses start, the assumptions to be used in the analyses will be defined. These assumptions include the following:

- The basis for which scenarios will be considered in the TSPA analyses for site recommendation and the LA
- The basis for which alternative conceptual models and parameters from the abstraction activities will be considered in the TSPA analyses for site recommendation and the LA
- The basis for which designs (both repository and waste package) will be considered in the TSPA analyses for site recommendation and the LA
- The basis for refining the base case from which all sensitivities will be conducted

These assumptions will define the range of analyses that will be conducted for the TSPA for the site recommendation and LA. The base case will be constructed from a composite of the scenarios refined from the TSPA-VA. It will include the base hydrologic scenarios and some perturbations on the base case (e.g., climate change). The base case will attempt to incorporate alternative conceptual models for degradation of the waste package, dissolution of the waste form, transport through the near field and the engineered barrier system, and aqueous transport in the unsaturated and saturated zones.

The models for TSPA analyses will be derived from the abstractions of the process models. Stochastic techniques will be used for the analyses to produce results based on a variety of model inputs. If required, alternative numerical representations of processes will be used. The results will be presented as distribution functions showing probability versus effect (e.g., dose, release, or other risk measures). The activities required in this effort include the following:

- Integrate all features, processes, and events developed in various abstraction activities into a defensible representation of those features, processes, and events that are most important.
- Combine all relevant abstracted model results, relevant simplified process models, relevant design information, and TSPA-specific assumptions into the selected TSPA software.
- Determine performance for the base case set of parameters and evaluate the impact of alternative probability distribution function shapes.
- Review and revise the base case results, as appropriate.

Once the base case has been completed satisfactorily, a range of sensitivity and uncertainty analyses will be conducted to evaluate the impact of the ranges of parameter uncertainty, conceptual uncertainty, and scenario uncertainty. The impact on design alternatives will also be evaluated. The required activities include the following:

- Generate complementary cumulative distribution functions, which define the probability of the resulting performance, to evaluate parameter uncertainty. Complementary cumulative distribution functions will not be generated for all conceptual and scenario uncertainties or design alternatives; however, a select group focusing on the most significant processes or uncertainties will be evaluated.
- Analyze process sensitivity to define the key parameters that impact the total system performance.

- Use complementary cumulative distribution functions to evaluate uncertainty in those cases where parameter sensitivity is not evaluated explicitly. This approach uses "expected value" predictions.

- Review the sensitivity analyses with the site and design organizations before the analyses are documented in the TSPA for the site recommendation and the LA to identify key uncertainties in site and design-related processes, models, and parameters that significantly impact performance.

Prioritize the focus of the sensitivity analyses. This focus will be based on the importance of the model as identified in the TSPA for the site recommendation and the LA and the repository safety strategy available at the time of the analyses.

Results of the TSPA for the site recommendation and the LA to assess compliance of the potential repository system at Yucca Mountain will be interpreted and documented. This activity will include the following steps:

- Document the base case assessment of postclosure performance in sufficient detail to understand how all the elements in the TSPA analysis have been combined. Document the full suite of sensitivity analyses so the significance of alternative assumptions can be evaluated to increase confidence in the overall analysis.

- Choose a reasonably conservative reference case.

- Conduct sensitivity analyses to describe the impact of the conservative assumptions.

- Investigate possible low-likelihood adverse models, or parameters, to illustrate the potential impact of such unlikely assumptions on performance.

- Describe the implications of these results for performance confirmation and design and recommend testing and design modifications that could reduce uncertainty or enhance performance.

- Complete a technical, regulatory, and management review of the TSPA for the site recommendation and LA document to ensure its suitability for licensing and revise the document, as appropriate.

Using appropriate degrees of detail and illustration, summary documents for scientific and public audiences will be developed. Visualization techniques for displaying the analysis and results on the internet, including hypertext versions of regulatory, scientific community, and public documents will also be developed.

The completion of TSPA for the site recommendation and LA will support addressing the NRC key technical uncertainty on the TSPA and technical integration by documenting the methodology for implementing TPSA analyses and documenting the TSPA analyses. This work specifically addresses the following third subissue of this key technical issue: Are the major components of the DOE TSPA methodology (i.e., model abstractions; probability and consequences of the relevant features, events, and processes; parameter and model uncertainties; and bounding assumptions) sufficiently comprehensive to provide a defensible safety case (Section 4.3.3.7)? These activities (2396-2399) address the first element of the postclosure safety case (assessment of the expected postclosure performance) by documenting the expected behavior of the repository system. Sensitivity studies and uncertainty analyses will address the second element of the postclosure safety case (defense in depth and design margin).

This work will support Milestones M2JH, TSPA-SR/LA Methodology and Assumptions Documents; M2JJ, TSPA-SR/LA Base Case Results Document; and M2JG, TSPA-SR/LA Documentation.

3.3.3 Design Support

This work will continue the quantitative evaluation of design alternatives, features, and options to support development of the technical basis for selecting the initial site recommendation and LA design. The evaluations will build on the TSPA-VA analyses of the VA reference design, design options and design alternatives that were evaluated in TSPA analyses supporting the draft environmental impact statement. The focus of the work will be to evaluate a reasonable range of uncertainty in the design features included in the options and alternatives study and to couple this uncertainty analysis with reasonably bounded uncertainty in the natural system that may be compensated by these design enhancements. This will require additional abstractions of process models from the scientific program and from studies of the waste package and engineered barrier system. These analyses contribute to the evaluation of defense in depth and design margin.

The analysis will focus on key functions of the repository system; keeping water off the waste packages, increasing waste package lifetime, limiting the rate of release of radionuclides from breached waste packages, and reducing concentrations of radionuclides in groundwater during transport away from the waste packages. This activity (2185) provides sensitivity and uncertainty analyses that address the second element of the postclosure safety case (defense in depth and design margin).

This work supports Milestone M2MQ, Select Initial SR/LA Design and Options.

4. STATUTORY AND REGULATORY ACTIVITIES

In addition to the technical activities required to support testing, design, and performance assessment, a substantial body of other work is needed to comply with statutory and regulatory requirements. This section summarizes the other statutory and regulatory work needed between the VA and submittal of the LA. This section discusses the statutory and regulatory framework for site recommendation and submittal of an LA, and describes the activities and documentation that must be completed to achieve these milestones. The development of an EIS is included. The schedule for these activities is in Section 7, and the cost of these activities is in Section 6. The technical work described in Sections 2 and 3 is intended to provide information and documentation that will be sufficient to complete the site recommendation and the preparation of an LA that will be acceptable for NRC review.

4.1 ENVIRONMENTAL IMPACT STATEMENT AND ENVIRONMENTAL COMPLIANCE

The work scope focuses on preparing the EIS and maintaining a comprehensive environmental compliance program for all other activities. The environmental program ensures that every aspect of the project follows state and federal rules to protect public health and the natural surroundings of Yucca Mountain.

4.1.1 Environmental Impact Statement

As discussed in Section 1.9 of Volume 1, if the Yucca Mountain site is found suitable, the Secretary of Energy may then recommend that the President of the United States approve the site for developing a repository. An EIS must accompany such a recommendation. The EIS will follow criteria established by the Nuclear Waste Policy Act (1989) and the National Environmental Policy Act of 1969, and rules, or implementing regulations, issued by the Council on Environmental Quality. The EIS also will follow the DOE implementing regulations and orders.

The EIS will evaluate the potential environmental impacts associated with constructing, operating and monitoring, and eventually closing a repository at Yucca Mountain as the proposed action. Under Section 114 (a)(1)(D) of the NWPA, the EIS is not required to consider alternatives to geologic disposal, need for a repository, or alternatives to a repository. The EIS will evaluate a proposed repository designed to dispose up to 70,000 metric tons of spent nuclear fuel from commercial reactors, and DOE-owned spent nuclear fuel and high-level radioactive waste. Based on public comments on the scope of the EIS, the document will also include analyses for disposing of all projected spent nuclear fuel and other highly radioactive waste types that may be appropriate for disposal at Yucca Mountain. Public comments were received during public scoping meetings.

The proposed action evaluates potential environmental impacts associated with developing a repository, as well as transporting wastes to the repository. DOE has developed implementing alternatives to evaluate the range of potential environmental impacts. The implementing alternatives are defined to include a reasonable range of expected activities based on different approaches DOE might use to implement the proposed action.

DOE has developed three different design concepts (referred to as implementing design alternatives) to be evaluated in the EIS. These concepts are intended to reasonably estimate the potential impacts of the proposed repository. The concepts are based on thermal loading, that is, how much heat the waste packages could generate once inside the repository and how this heat could be distributed. Each concept would reflect differences in the size of the subsurface repository, as well as the layout or configuration of drifts and the spacing between them. The expected range of environmental impacts would be reflected by the differences in design. The range of conditions represented by the different concepts also could affect the long-term repository performance. The EIS will also examine the potential differences in environmental impacts that might occur as a result of variations in the design concepts. These variations represent the range of alternative designs DOE is considering.

Evaluation of the potential differences in impact caused by changes in design will ensure that the three concepts reasonably address a full range of potential impacts associated with the proposed action.

Various transportation choices within Nevada will be evaluated for shipping spent nuclear fuel and high-level radioactive waste to the repository. These choices are more correctly referred to as rail/intermodal implementing alternatives. The choices consider routes and modes of delivery: train and trucks. The radioactive materials could travel to the proposed repository using either a single mode of transportation or a combination of modes. Packaging options include evaluating canistered and uncanistered spend nuclear fuel.

Five potential rail corridors within the State of Nevada will be evaluated, as will three potential intermodal transfer stations that would receive wastes from outside the state. Two national transportation options will also be considered. The first involves shipping the majority of spent nuclear fuel and high-level radioactive waste by legal-weight truck from the site where the waste originated to the repository. Under the second option, the majority of shipments would be made by rail, except for shipments from those sites without the ability to handle large transportation casks.

The draft EIS will be completed and issued to the public for review and comment in July 1999. All comments will be reviewed and categorized according to subject matter and issues raised. DOE will then develop responses to the public's comments. These responses will be published as part of the final EIS. DOE will review the discussions in the draft EIS to decide whether revisions to the EIS should be made based on the comments received. DOE will publish the EIS in August 2000 as discussed in Section 7.

As required by the Nuclear Waste Policy Act, the final EIS will accompany the DOE recommendation to the President that the Yucca Mountain site be developed as a repository. The EIS also will be a part of the DOE license application to NRC for authorization to construct a repository. The Nuclear Waste Policy Act directs NRC to adopt the EIS to the extent practicable. DOE is coordinating with NRC regarding the EIS. Consistent with its authority, NRC will participate as a commenting agency in the DOE EIS process for the repository.

4.1.2 Environmental Compliance

The Nuclear Waste Policy Act, other applicable laws, and DOE directives (orders, policies, standards, regulations and other implementing documents) require compliance with environmental regulations to protect public health and the natural surroundings of Yucca Mountain. An environmental compliance program has been implemented to ensure that project activities are planned and conducted to meet the applicable requirements.

Environmental compliance is a dynamic process and covers all the work planned for the site, from characterization to the eventual design, construction, operation, closure, and decommissioning of the proposed Monitored Geologic Repository. The scope of work that the process covers is extensive. Work begins by identifying applicable requirements and ensuring that new regulations or modifications to existing regulations are linked to planned activities and data needs. Programs are in place to ensure that critical issues are addressed. These programs cover the following:

- Land use, including reclamation
- Biological resources
- Air quality and weather (meteorology)
- Water quality and availability
- Cultural resources and values, with emphasis on the concerns of Native Americans
- Background and natural radiation monitoring
- Environmental justice
- Hazardous and solid wastes and materials

The compliance process oversees how the project prepares, modifies, and submits applications for federal and state permits. The permits authorize various activities at the site, such as testing and construction. Permits for the proposed repository will be identified and obtained, as needed to support work, between completion of the VA and NRC authorization to construct the repository. After the permits have been issued, monitoring will continue to ensure that all permit objectives are met and compliance is maintained. This oversight process will include formal audits. The audits comprise inspections, surveillance tasks, and assessments to ensure that project activities are performed in accordance with applicable environmental laws, regulations, and permits.

The environmental compliance process also evaluates land ownership and withdrawal actions to support the LA. To meet the requirements of 10 CFR 60.121, DOE must establish jurisdiction and control over the land on which a repository is located. The site of the proposed repository at Yucca Mountain is located on land owned by the federal government and under jurisdiction and control of the U.S. Department of Interior, Bureau of Land Management (BLM); U.S. Department of Defense, U.S. Air Force; and DOE. Land acquisition for DOE use is controlled by BLM regulations in 43 CFR 2300, unless withdrawal is made as directed by an act of Congress. DOE will pursue available administrative mechanisms for land acquisition including the filing of an application with BLM. Acquisition of land using administrative mechanisms would not accomplish a permanent land withdrawal, but could be used to support the initial licensing process until Congress has appropriate opportunity to act on permanent withdrawal legislation.

4.2 SITE RECOMMENDATION

If the Yucca Mountain site is determined to be suitable for location of a repository under the criteria developed under NWPA Section 113, Site Characterization, the Secretary of Energy may recommend that the President approve the site for development. NWPA Section 114, Site Approval and Construction Authorization, provides for a process for site recommendation and describes the documentary basis for this recommendation.

The site recommendation process begins with public hearings near the Yucca Mountain site for the purposes of informing residents of the area that the Secretary is considering recommending the site, and to receive the residents' comments regarding the possible recommendation. These hearings are referred to as "consideration hearings" and would be announced by a notice of consideration. At the time that this notice is published, DOE would also request the views and comments of the governors and legislatures of all of the states, and the governing bodies of any affected Indian tribes.

If, upon completion of these hearings and completion of site characterization, the Secretary decides to recommend approval of the site to the President, the Secretary will notify the governor and legislature of the State of Nevada. No sooner than 30 days later, the Secretary will submit to the President a recommendation that the President approve the site for the development of a repository.

Together with the site recommendation, the Secretary will make available to the public, and submit to the President, a comprehensive statement of the basis for this recommendation. The contents of this comprehensive site recommendation statement are described in NWPA Section 114. Figure 4-1 graphically depicts the bases for the statement. Table 4-1 quotes the NWPA content requirements and summarizes the documents and actions with which DOE plans to satisfy these requirements.

In addition to the DOE documentation, the site recommendation statement must include the preliminary comments of NRC concerning the extent to which the at-depth site characterization analysis and the waste form proposal seem to be sufficient for inclusion in the license application. The views and comments of the Governor and legislature of any state, or the governing body of any affected Indian tribe are also to be included, along with the Secretary's response to these views.

If the President considers the site qualified for application for a construction authorization, the

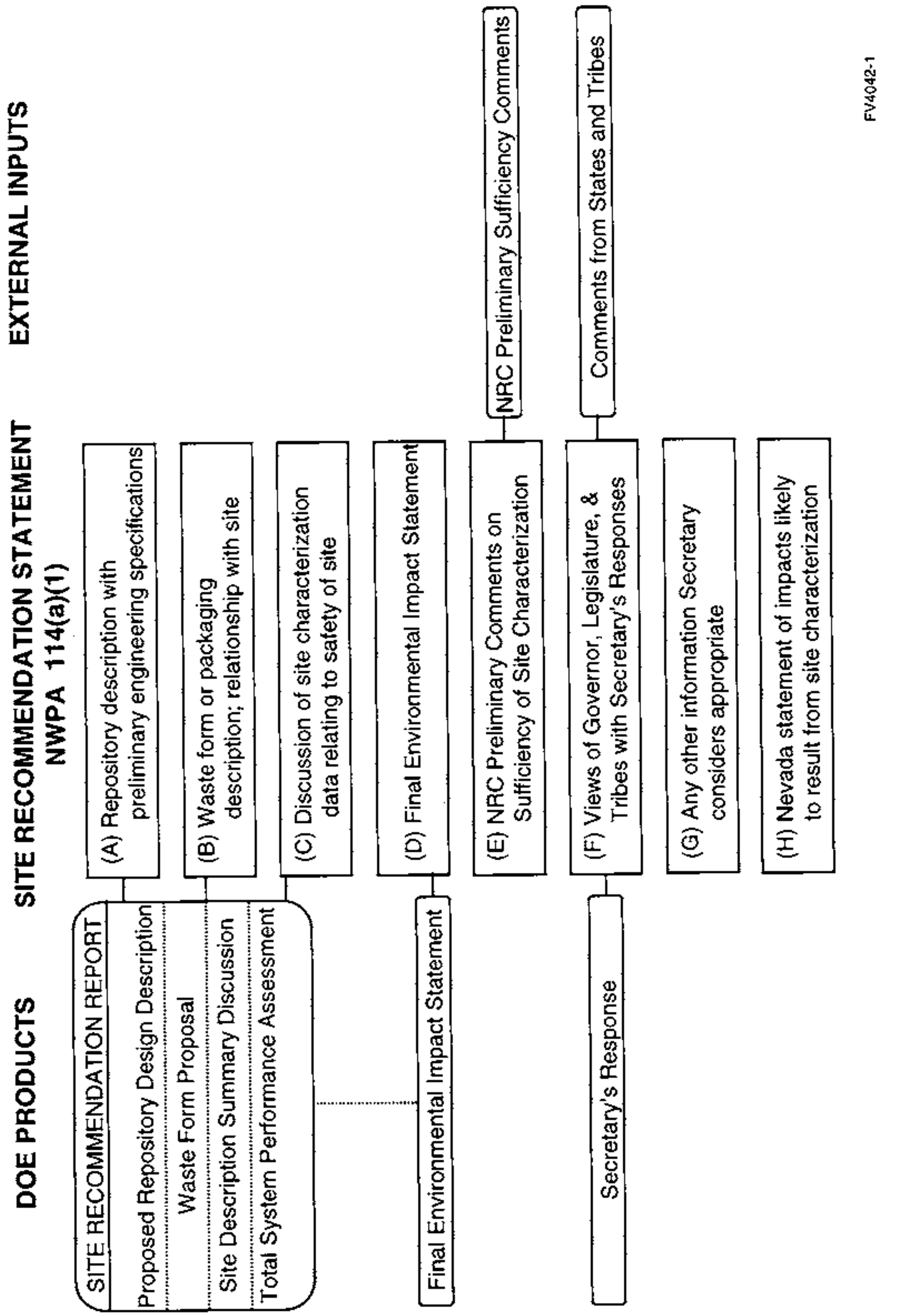

Figure 4-1. Bases for the Site Recommendation Statement

Table 4-1. Documents and Actions Planned to Satisfy Nuclear Waste Policy Act Requirements

Nuclear Waste Policy Act Requirements	Planned Documents and Actions to Satisfy Requirements
"a description of the proposed repository, including preliminary engineering specifications for the facility;"	The reference design description for a geologic repository and supporting information being developed in support of the LA design will provide the necessary input to adequately describe the proposed repository. In addition, the system description documents being produced to guide the design of the systems important to radiological safety and waste isolation will be used to provide the necessary input to describe the preliminary engineering specifications for the facility.
"a description of the waste form or packaging proposed for use at such repository, and an explanation of the relationship between such waste form or packaging and the geologic medium of such site;"	The reference design description, waste package design description, and the supporting technical documents developed for the production of the LA design will be used to provide a description of the waste form or packaging proposed for use at a repository. The site description and the TSPA will be used to develop the explanation of the relationship between the waste form or packaging and the geologic medium of the site.
"a discussion of data, obtained in site characterization activities, relating to the safety of such site;"	The site description, the TSPA, and supporting technical documents will be used to provide a discussion of data, obtained in site characterization activities that relate to the safety of the site.
"a final EIS prepared for the Yucca Mountain site pursuant to subsection (f) and the National Environmental Policy Act of 1969 [42 USC 4321 et seq.], together with comments made concerning such EIS by the Secretary of the Interior, the Council on Environmental Quality, the Administrator, and the Commission, except that the Secretary shall not be required in any such EIS to consider the need for a repository, the alternatives to geologic disposal, or alternative sites to the Yucca Mountain site;"	The final EIS, including the comments concerning the EIS will be included in an appendix.
"preliminary comments of the Commission concerning the extent to which the at-depth site characterization analysis and the waste form proposal for such site seem to be sufficient for inclusion in any application to be submitted by the Secretary for licensing of such site as a repository;"	In order to include the NRC preliminary sufficiency comments in the site recommendation report as an appendix, OCRWM must establish a basis for sufficiency with NRC through a series of interactions, sharing of data and analyses, as requested, and must work toward addressing the key technical issues identified by NRC. Products currently planned to be complete by the time of the request are expected to provide the basis for NRC's sufficiency determination. At the time the notice of consideration is published, OCRWM will request by letter that NRC initiate the development of the preliminary sufficiency comments. The letter will specifically request NRC to provide preliminary comments concerning the extent to which the at-depth characterization analysis and waste-form proposal for the proposed Yucca Mountain site seems to be sufficient for inclusion in a license application. The request will specify the date by which the NRC comments are needed for inclusion in the final site recommendation report.
"the views and comments of the Governor and legislature of any State, or the governing body of any affected Indian tribe, as determined by the Secretary, together with the response of the Secretary to such views;"	At the time the notice of consideration is published, the governors and legislatures of all 50 states, any affected Indian tribes and the affected units of local government will be requested by letter to provide their views and comments on the recommendation of the Yucca Mountain site as a repository. Any comments and views received from the governor or legislature of any state will be reviewed and responses will be developed. These comments and responses will be included as an appendix.
"such other information as the Secretary considers appropriate;"	Updated cost estimates and transcripts from the consideration hearings may be among the other information that the Secretary may consider appropriate for inclusion.
"and any impact report submitted under section 116(c)(2)(B) [42 USC 10136(c)(2)(B)] by the State of Nevada." (Per Section 116 of NWPAA [Nuclear Waste Policy Act as amended] any such impact report would accompany a request for financial or technical assistance and consist of: "...a report on the economic, social, public health and safety, and environmental impacts that are likely to result from site characterization activities at the Yucca Mountain site. Such report shall be submitted to the Secretary after the Secretary has submitted to the State a general plan for site characterization activities under section 113(b)")	Any such request or report will be included or attached to the site recommendation. However, no such request or report has been received.

Source: Nuclear Waste Policy Act of 1982.

President will submit a recommendation of the site to Congress. Under NWPA Section 116, Participation of States, the State of Nevada has 60 days to submit a notice of disapproval to the Congress. This notice must be accompanied by a statement of reasons why the state disapproved the recommended site.

In order to complete the recommendation process, Congress must pass a resolution of repository siting approval during the first period of 90 calendar days of continuous session. If the state does not disapprove, or if Congress passes an approval resolution, the site is designated and DOE must submit an application for construction authorization to NRC within 90 days.

4.3 LICENSING

This section describes the licensing activities that lead up to, and directly support, developing the LA.

4.3.1 Licensing Activities

Several activities are planned to prepare the LA. These activities fall into five functional areas. The scope of work involves the following:

- Preparing licensing documents
- Resolving outstanding issues
- Managing technical data
- Managing records
- Integrating electronic information

4.3.1.1 Licensing Documents

Five planned DOE documents specifically related to the licensing of Yucca Mountain as a repository include the LA for construction authorization and four documents that will be useful in the preparation of the LA (the technical guidance document, an initial and updated licensing case selection report, the final report of a series of three seismic topical reports, and the disposal criticality topical report). A documentary record will identify all licensing basis documentation and will provide traceability to the source of documentation. Regulations will continue to be reviewed often for applicability to the site characterization project.

The LA will address applicable NRC requirements and key technical issues and will reflect NRC regulatory requirements. The LA will present information on the site, repository design, and performance assessment to support the licensing safety case. Preparation of the LA will follow the *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain* (YMP 1997a).

A DOE technical guidance document currently under development will apply the applicable NRC requirements and guidance for preparation of the LA. This documented guidance will define the format and content, based on current NRC regulations, nuclear regulatory guides, informal prelicensing advice, and applicable nuclear industry standards. The document will contain acceptance criteria for information that must be included in the LA, with enough detail to support the licensing safety case. NRC has provided issue resolution status reports containing acceptance criteria that will be part of the NRC review of the LA. The technical guidance document will be revised as necessary to provide sufficient guidance for all sections of the LA and reflect current regulations.

A report will be developed and updated (see milestone schedule in Section 7) to document the basis for the licensing safety case (i.e., preclosure and postclosure safety cases as discussed in Section 2) for design and performance assessment at site recommendation and LA submittal. The proposed licensing safety case will define performance allocation, design margin, use of governing codes and standards, documentation of the proposed defense in depth implementation, identification of the proposed performance confirmation scope and expectations, and identification of any program enhancements addressed by the design. These decision points are part of the iterative process necessary to develop the LA design and the TSPA for the LA and to demonstrate compliance with NRC requirements in the LA.

The final of three seismic topical reports and the disposal criticality topical report are planned to identify the methods used for the licensing safety case. These reports will be submitted to NRC to determine the acceptability of the DOE proposed methods. Requests by NRC for clarification or additional information in the form of topical report amendments or revisions would be expected for NRC to determine the acceptability of the proposed methods. Documentation of NRC approval would be received in a prelicensing evaluation report, similar to a safety evaluation report in the nuclear power industry. The methods and information in the topical reports will be reflected in the LA. *Determination of Preclosure Seismic Design Basis for a Geologic Repository at Yucca Mountain* will be the last of three seismic reports. This report is intended to demonstrate that the seismic design bases address potential hazards from future underground atomic tests and from natural earthquakes. The disposal criticality topical report will present the methods for evaluating and minimizing the probability and consequence of criticality during repository postclosure.

The LA will be developed in two phases: a working draft and an acceptance draft. The acceptance draft LA will provide more complete information than the working draft LA and will incorporate revisions based on feedback from the internal DOE review of the working draft. The acceptance draft will likely reflect revisions based on continuing prelicensing interactions with NRC, including information to address NRC key technical issues and NRC acceptance criteria, information reflecting agreements about the level of detail to be presented in the LA, and information provided in response to relevant NRC guidance that may be developed following the submittal of the VA. After an agency-wide review, and assuming that the site is recommended and the recommendation approved, the completed LA will be submitted to NRC for docketing and review. The schedule of milestones for LA development is included in Section 7.

A preliminary emergency plan will be prepared to serve as the basis for the presentation of emergency planning contained in the LA. The plan will present preparations made and planned to cope with preclosure radiological emergencies and comply with then-current regulatory requirements. The emergency plan will present planned organizational structures, staffing, administrative controls, facilities, and interface communication with response organizations. Also included will be descriptions of accident level conditions and mitigating actions to manage the potential consequences of an accident to minimize its effect.

The documentary record will provide a complete listing of the information used to prepare the LA. The record will also provide traceability to the information source in the records and technical information systems, the technical data management system, the three-dimensional model warehouse, and the controlled design assumptions by tracking number. The documentary record will consist of referenced citations, reports, evaluations, models, calculations, design packages, and technical data. The complete record will also contain a history of the development of supporting documents for site recommendation, including all information, assumptions, and rationale. Additionally, the documentary record will contain documentation of key decisions important to the licensing process, that is, decisions having significant effects on site characterization; technical performance, schedule and/or cost of the design, construction, or operation phases of the repository; program policy; or external project interactions. A technical database will identify and track the technical data used to develop the site description, the engineered barrier system and repository designs, and the performance assessments.

4.3.1.2 Issue Resolution

During site characterization, DOE has worked to understand potential licensing issues and determine the information needed for resolution. NRC has refocused its program around 10 key technical issues deemed most important to repository performance (see Section 4.3.3). Issue resolution status reports developed by NRC provide technical acceptance criteria for the key technical issues against which the DOE LA will be reviewed and accepted. Through these reports, NRC provides

DOE feedback on how to resolve these issues during the prelicensing consultation period; however, any such resolution would not preclude NRC from raising and considering these issues during the licensing proceedings. Work performed between the VA and LA submittal will focus, in part, on resolving as many of these issues as feasible. Significant issues remaining at the time of LA submittal are to be clearly described in the LA, with a schedule for their resolution.

DOE will attempt to address key technical issues and other issues in a variety of ways, based on acceptance criteria from the issue resolution status reports. These methods of addressing the issues include completing topical reports that are now in preparation, interactions, official correspondence, NRC comments and responses, and review of regulations as they apply to the licensing documents. These activities have five goals:

- Obtain NRC acceptance during the prelicensing consultation for new or different methodologies for which there is no licensing precedent.
- Revise or clarify NRC requirements.
- Resolve technical and regulatory issues.
- Address NRC comments on technical documents or activities.
- Identify, track, and complete commitments to NRC.

DOE will continue to use topical reports for seismic design methodologies and disposal criticality analysis methodologies as a tool to address these issues with NRC: DOE and NRC have agreed that these issues are appropriate subjects for such reports. The seismic design topical report will be the final of three seismic topical reports. It will support the preclosure safety case, safety classification and design basis events, and help address the key technical issue, Structural Deformation and Seismicity. The disposal criticality topical report will support the postclosure safety case for the analysis of postulated disruptive events of nuclear criticality and will help address the key technical issues Evolution of the Near-Field Environment and Container Life and Source Term. When completed, these topical reports should provide the bases for NRC prelicensing evaluation reports.

DOE will continue identifying, tracking, and completing commitments to NRC, as well as tracking and closing open items from *Site Characterization Plan: Yucca Mountain Site, Nevada Research and Development Area* (DOE 1988).

4.3.1.3 Technical Data Management

Between completion of the VA and LA submittal, DOE will continue to manage technical data in technical databases, managed databases, and data sets generated and used by DOE, and maintain data quality for use in supporting the LA safety case. Data are qualified through data qualification options identified in *Quality Assurance Requirements and Description for the Civilian Waste Management Program* (DOE 1998a), which will be used in support of the LA. Requirements for managing technical data come from *Technical Data Management Plan* (YMP 1997b), 10 CFR 60, *Processing of Technical Data on the Yucca Mountain Site Characterization Project* (YMP 1996), and the NRC-approved *Quality Assurance Requirements and Description.*

Technical databases contain the following:

- Spatial data providing location information for test activities, facilities, roads, and geoscience features within the Yucca Mountain study area
- Summaries and interpretations of experts' knowledge about site and regional geocharacteristics data and selected engineering test data
- Data from site investigation field and laboratory tests and engineering analysis input

- Data on waste inventories to be received at the repository

A technical database for design requirements, technical basis, and configuration is planned as well. The data in such a database would be used to support the LA.

Managed databases include characteristics of radioactive waste materials, traceability of requirements imposed on the program and their implementation, and chemical and thermodynamic properties of chemical species. These databases facilitate repository system design and performance assessment evaluations.

Data sets are groups of data used for specific purposes, such as design input, EIS, LA, or performance assessment. These data sets are used to develop key reports on the suitability of the repository, its environmental impact, and associated licensing activities and evaluations.

A master indexing system provides traceability for the data from its original source to the highest level of development and interpretation, using data tracking numbers.

4.3.1.4 Records Management

The records information system will be upgraded to improve the capability to search and retrieve records, and to address changes in information management regulations and user needs. In addition, records processed before fiscal year 1996 are now being converted to be computer searchable to allow the identification of records and the computer retrieval of the image and text of all licensing documents.

4.3.1.5 Licensing System Network

An integrated electronic information system will be developed that will consist of the NRC electronic docket and the DOE system. This system, resulting from discussions with NRC and in anticipation of a revision to 10 CFR 2, Subpart J, will use available internet technology for designing and developing an electronic information management and distribution system rather than the licensing support system concept in the current version of 10 CFR 2, Subpart J. Technology has advanced rapidly since the licensing support system concept was developed. DOE and NRC acknowledge that an electronic docket in an internet-compatible format would best serve their current needs. NRC is revising 10 CFR 2, Subpart J to reflect the change in concept and DOE will comply with the final regulation. DOE will conduct certification reviews of its portion of the integrated electronic information system and associated documentary material for compliance with the proposed revision to 10 CFR 2, Subpart J (see milestone schedule in Section 7).

The new system concept will allow access from the NRC electronic data system to the DOE records management system, the online LA, and the technical database management system only through the DOE Table of Contents. The DOE Table of Contents will guide users to licensing documents placed on the internet's World Wide Web, with hypertext links to reference documents in the records management system and source data in the technical data management system.

When the LA is submitted to NRC, copies of the records supporting the LA and documents placed on the Internet without the hypertext links will be sent to NRC and become part of the NRC electronic docket.

4.3.2 License Application Status and Schedule

Detailed plans were made in fiscal year 1998 to start preparing a working draft LA to support the site recommendation process and the LA promptly after completion of the VA. These plans include a schedule for writing sections of the LA and a detailed cost estimate for developing the working draft LA and the LA. This section briefly discusses the schedule for the LA and the activities that will help ensure its successful completion. Important activities and milestones supporting LA development are also shown in Section 7.

4.3.2.1 License Application Schedule

There are three major milestones for developing the LA:

- 1998
  Development of working draft commences

- 2001
  Acceptance draft complete

- 2002
  LA submitted to NRC (must be submitted within 90 days after the site recommendation becomes effective)

Developing the LA begins with creating a working draft. The working draft, to be completed in 1999, has three objectives:

- Develop the basic framework of the licensing case for each chapter and section of the LA.

- Help ensure that any information required for licensing but missing from the working draft will be in place when needed.

- Allow an interim assessment of progress in developing the final document.

DOE does not plan to rearrange project work priorities to have a complete licensing case in the working draft LA; consequently, some sections of the working draft are expected to be incomplete.

DOE will continue work on the LA while the working draft is being reviewed. The acceptance draft will be completed in 2001 to support the assumed date that the LA will be submitted to NRC for docketing (2002). The acceptance process will involve extensive DOE management, technical, and legal reviews.

4.3.2.2 Preparatory Activities

DOE developed two important documents in fiscal years 1997 and 1998 that will directly support the development of the LA: *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain* (YMP 1997a) and *Technical Guidance Document for License Application Preparation* (in prep.). The first document is a guide for managing the process to prepare the LA. The second publication provides technical guidance to the authors. This document will describe the types of information the LA will include. The technical guidance document also will provide acceptance criteria (including applicable criteria from the NRC issue resolution status reports, as applicable) to assist authors and reviewers in assessing the adequacy of the information provided in the LA. DOE will update this document in fiscal year 1999 to include additional acceptance criteria that will have been developed by that time. The document also will be updated to reflect any new site-specific regulations for Yucca Mountain that NRC may issue.

DOE also has summarized changes to activities planned in the site characterization plan. This summary, *Documentation of Program Change* (CRWMS M&O 1998c), includes changes in focus and studies that were added or deleted during site characterization. In contrast to the semiannual site characterization progress reports, which describe program changes occurring in the 6-month periods before their publication, the *Documentation of Program Change* summarizes changes made over the entire site characterization period. This publication, which is updated annually, will be an important source of information for changes that must be addressed in the LA.

4.3.3 Nuclear Regulatory Commission Interactions and Key Technical Issues

DOE has taken many steps to address licensing issues during site characterization, including meetings with NRC to focus on specific concerns and development of topical reports. Periodic management meetings have been held between DOE and NRC to further facilitate communication on, and resolution of, licensing issues. The meetings, like all planned meetings between DOE and NRC to discuss the repository program, have been, and will continue to be, open to the public. External stakeholders have been invited to attend.

DOE has submitted three topical reports to allow early NRC review of DOE proposals to address certain important technical issues:

- *Evaluation of the Potentially Adverse Condition "Evidence of Extreme Erosion During the Quaternary Period" at Yucca Mountain, Nevada* (DOE 1993)

- *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (DOE 1997a)

- *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain* (DOE 1997b)

After reviewing the DOE topical report on extreme erosion and supplemental information, NRC had no further questions on this issue. Progress has also been made in addressing the seismic issues identified in the last two topical reports.

DOE is writing two more topical reports; one on the preclosure seismic design basis for a geologic repository at Yucca Mountain, and one on methods to analyze disposal criticality. Both of these reports will be completed during fiscal year 1999.

Early in 1995, NRC staff recognized the need to refocus its prelicensing repository program on resolving issues most significant to repository performance. From a licensing perspective, the term "resolved" means that NRC has no more questions at the present time; however, the issue may be reopened later if warranted by new information, different interpretations, or other considerations. Two events provided a technical basis for the NRC refocusing effort: a reorganization of the DOE high-level radioactive waste work into a structure identified as the program approach, and a report issued to EPA by the National Academy of Sciences that recommended establishing a radiation protection standard for a proposed high-level radioactive waste repository sited at Yucca Mountain. The report by the National Academy of Sciences was mandated by Section 801 of the Energy Policy Act of 1992. The Act also directed EPA to develop a standard for the site-specific, health-based radiation protection standard for Yucca Mountain. The standard must be consistent with the recommendations of the National Academy of Sciences. The act further stipulates that NRC must modify its technical requirements and criteria (described in the disposal regulations, [10 CFR 60]) to be consistent with the standard.

In 1996, NRC identified 10 key technical issues (NRC 1997b-g; 1998a-c) intended to reflect the topics that NRC considers most important to performance of a high-level nuclear waste repository at Yucca Mountain:

- Igneous Activity (Section 4.3.3.1)

- Structural Deformation and Seismicity (Section 4.3.3.2)

- Evolution of the Near-Field Environment (Section 4.3.3.3)

- Container Life and Source Term (Section 4.3.3.4)

- Thermal Effects on Flow (Section 4.3.3.5)

- Repository Design and Thermal-Mechanical Effects (Section 4.3.3.6)

- Total System Performance Assessment and Integration (Section 4.3.3.7)

- Activities Related to the Development of the EPA Yucca Mountain Standard (Section 4.3.3.8)

- Unsaturated and Saturated Flow Under Isothermal Conditions (Section 4.3.3.9)

- Radionuclide Transport (Section 4.3.3.10)

The issues provide the basis for focusing NRC evaluations and developing an independent understanding of the issues and their relative importance to repository system performance. Various combinations of the key technical issues include all of the principal factors that support the repository safety strategy (DOE 1998b). The NRC staff is developing issue resolution status reports for the key technical issues. The reports provide information on the resolution status of the key technical issues and associated open items related to the site characterization analysis (NRC 1989) and provide timely feedback to DOE on efforts to address the issues. Information on status of nine of the key technical issues and identification of work planned by DOE to address these nine issues are summarized in Sections 4.3.3.1 through 4.3.3.10. NRC has indicated that there are no plans to provide an issue resolution status report for the key technical issue, Activities Related to the Development of the U.S. Environmental Protection Agency Yucca Mountain Standard.

Key technical issues are being addressed at the NRC staff level, in advance of an LA submittal, through various interactions between DOE and NRC. The NRC staff describes progress in addressing and resolving the key technical issues in a series of periodic issue resolution status reports (NRC 1997b-g and 1998a-c), which document significant progress toward issue resolution and provide timely feedback to DOE regarding specific issues or subissues. To date, NRC has made significant progress in developing the first iteration of the issue resolution status reports; these reports address parts of selected subissues of eight of the ten key technical issues. NRC is preparing an issue resolution status report on the key technical issue of radionuclide transport, and NRC has planned to prepare revisions of the issued reports to address subissues not previously addressed.

The issue resolution status reports also identify acceptance criteria that NRC staff may use to review and evaluate the adequacy of information related to the key technical issues. DOE intends to use the issue resolution status reports as part of the basis to evaluate the adequacy of site characterization, design, and performance assessment activities, and prioritize planned work. References to discussions in other volumes of the VA related to the key technical issues are provided in Table 4-2.

DOE interactions with NRC between now and submittal of the LA will focus on resolving key technical issues and reaching a common understanding about content of the LA. These interac-

Table 4-2. Locations of Information Related to Key Technical Issues in Viability Assessment Volumes 1 Through 4

Key Technical Issue	Volume 1 Sections	Volume 2 Sections	Volume 3 Sections	Volume 4 Sections
Igneous Activity	2.2.7		2.3.2, 4.4.2	3.1.1, 3.3.1
Structural Deformation and Seismicity	2.2.1, 2.2.7		3.4.1, 4.4, 4.4.3	3.1.1, 3.2.1, 3.2.2, 3.2.3, 3.3.1, 4.3.1
Evolution of Near-Field Environment	2.2.6	5.1.3	2.2.3, 3.4.2, 4.1.3, 4.1.4, 5.2.2, 6.5.1	3.1.2, 3.1.4, 3.1.5, 3.2.1, 3.3.1
Container Life and Source Term		5.1.3, 5.1.4	2.2.3, 3.2.2, 3.3, 3.4, 3.5.2, 5.5.1	3.1.4, 3.1.5, 3.2.2, 3.3.1
Thermal Effects on Flow	2.2.6		3.1.1, 3.2, 4.1.5, 5.2	3.1.1, 3.1.2, 3.1.4, 3.1.5, 3.2.1, 3.3.1
Repository Design and Thermal-Mechanical Effects	2.2.6	3.1.2, 3.3.4, 4.2.2, 4.2.7, 4.3	5.2.2, 6.5.1	3.1.1, 3.1.4, 3.1.5, 3.2.1, 3.2.2
Total System Performance Assessment and Integration			2.1, 2.3.3, 3.5, 3.6, 3.7, 3.8, 3.8.3, 4.4, 6.5.3	3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.5, 3.2.2, 3.3.1
Activities Related to Development of the EPA Yucca Mountain Standard[1]	Not Addressed	Not Addressed	Not Addressed	Not Addressed
Unsaturated and Saturated Zone Flow Under Isothermal Conditions	2.2.2, 2.2.3, 2.2.4, 2.2.5		3.1.1, 3.2.3, 3.3.1, 3.4.2, 3.6.1, 3.7.2, 4.1.3, 4.1.4	3.1.1, 3.1.2, 3.1.3, 3.3.1
Radionuclide Transport[2]	2.2.5, 2.2.6		5, 5.2.2, 6.5.1	3.1.1, 3.1.2, 3.1.3, 3.1.4, 3.1.5, 3.2.2, 3.3.1

[1]NRC has indicated no issue resolution status report will be developed for this key technical issue.

[2]Issue resolution status report is not yet available. Subissue statements are expected to change when issue resolution status report is issued.

tions help to develop an understanding of the important issues to be resolved and provide a forum for reaching agreement on the technical approach and resolution of issues. The interactions are intended to identify and address those key technical issues whose resolution is essential to a successful LA. Information needed to address the key technical issues (including resolution status) will be described in the technical guidance document for development of the LA and included in the LA working draft (see Section 4.3.1.1).

The issue resolution status reports, along with frequent DOE and NRC interactions, have been helpful in identifying technical issues that can be resolved and those that require further information in order to reach resolution. Communications between DOE and NRC are discussed further in Section 4.3.3.11.

4.3.3.1 Summary of Igneous Activity Key Technical Issue

The igneous activity key technical issue has been defined by NRC staff as "predicting the consequence and probability of igneous activity affecting the repository in relationship to the overall system performance objective" (NRC 1998b). NRC considers the igneous activity issue important because of the potential for direct releases of waste material into the accessible environment, and the potential to alter groundwater flow paths and initiate mechanical, chemical, and thermal effects that could cause degradation of the waste package and other engineered barriers. This key technical issue comprises two subissues identified in the section that follows.

Objective of Key Technical Issue. The main objective of the igneous activity key technical issue is to evaluate the significance of igneous activity to repository performance by reviewing and independently confirming important data, and evaluating and developing alternative conceptual models for estimating the probability of occurrence and consequence of igneous activity at the proposed repository site.

Probability of Igneous Activity Subissue Status. The subissue related to the probability of igneous activity comprises three technical components:

- Definition of igneous events
- Determination of recurrence rates
- Examination of geologic factors that control the timing and location of igneous activity

The emphasis of this subissue is evaluating the probability of disruption of the repository by volcanic activity.

The NRC staff has indicated that the probability subissue has been resolved based on agreement on reasonable mechanisms and realistic ranges of critical parameters necessary to evaluate the likelihood and character of future igneous activity at or near the proposed repository site.

As discussed in Section 2.2.1.3, DOE considers that current information is sufficient to address the issue of igneous activity. The remaining work involves monitoring the site for indications of igneous activity, and refining conceptual models of igneous activity and incorporating igneous activity into disruptive events scenarios for consideration in TSPA. This work is identified in Sections 3.1.1 and 3.3.1.

Consequences of Igneous Activity Subissue Status. The second subissue considers the consequences of igneous activity that could occur within the repository and comprises five technical components:

- Definition of the physical characteristics of igneous events
- Determination of the eruption characteristics for modern and ancient basaltic igneous features in the Yucca Mountain Region and analogous geologic settings
- Models of the effect of the geologic repository setting on igneous processes

- Evaluation of waste and repository characteristics with regard to behavior during igneous events

- Determination of geologic system characteristics relevant to the probability and consequences of igneous activity

This subissue will be addressed by NRC in Revision 1 of the issue resolution status report.

The remaining work involves refining conceptual models of igneous activity and potential effects of igneous activity on the repository system, and incorporating igneous activity into disruptive event scenarios for consideration in TSPA. This work is identified in Sections 3.1.1 and 3.3.1.

4.3.3.2 Summary of Structural Deformation and Seismicity Key Technical Issue

This key technical issue (NRC 1997f) is intended to ensure that structural deformation and seismicity that may significantly affect the performance of a repository sited at Yucca Mountain have been identified and adequately characterized. This key technical issue comprises four subissues identified in the paragraphs that follow.

Objective of the Key Technical Issue. The primary objective of the key technical issue is to ensure evaluation of all aspects of the seismotectonic features, events and processes of the geologic setting of Yucca Mountain that have the potential to compromise the performance of the proposed repository.

Fault Slip Subissue Status: What are the viable models of faults and fault displacements at Yucca Mountain? This subissue was partially addressed in Revision 0 of the issue resolution status report on structural deformation and seismicity. NRC staff concluded that resolution has been achieved on the identification of faults that may significantly affect repository design or performance. The staff indicated that about 84 Type I faults have been identified near Yucca Mountain. Most of the data related to these areas have been collected. Studies of past seismic activity have quantified the likelihood of future seismic events. The remaining work involves monitoring mapped faults for new displacements, refining conceptual models of faulting and fault displacement at the site, refining the geologic framework model, and ensuring that new data are provided to design and performance assessment activities.

Site characterization work planned to address structural deformation and seismicity is identified in Section 3.1.1. Subsurface design activities related to seismicity are identified in Section 3.2.1, and waste package design activities related to seismicity are identified in Section 3.2.2. Performance assessment work related to seismicity is identified in the description of disruptive events in Section 3.3.1.

Seismic Motion Subissue Status: What are the viable models of seismic sources and seismic motion at Yucca Mountain? This subissue was not addressed in Revision 0 of the issue resolution status report on structural deformation and seismicity. However, NRC staff concluded that resolution has been achieved on one comment concerning site motion that was in the site characterization analysis (NRC 1989, NUREG-1347, Comment 66).

As discussed in Section 2.2.1.3, DOE considers that current information is sufficient to address this subissue. Studies of past seismic activity have quantified the likelihood of future seismic events and provided an upper bound to the mean ground acceleration at the site. The remaining work involves monitoring for indications of new seismic activity, refining conceptual models of ground motion at the site, refining conceptual models of seismic sources, and ensuring that new data are provided to design and performance assessment.

Site characterization work planned to address structural deformation and seismicity is identified in Section 3.1.1. Subsurface design activities related to seismicity are identified in Section 3.2.1, and waste package design activities related to seismicity are identified in Section 3.2.2. Surface design work related to seismic motion is identified in Section 3.2.3. Performance assessment work related to seismicity is identified in the description of disruptive events in Section 3.3.1.

Fractures and Site Discontinuities Subissue Status: What are the viable models of fractures and site discontinuity features at Yucca Mountain? This subissue was not addressed in Revision 0 of the issue resolution status report on structural deformation and seismicity. The staff concluded that adequate methods are currently available to reasonably estimate future site behavior with regard to fracturing.

As discussed in Section 3.1.1 of Volume 1, work on fractures and site discontinuities is nearly complete. Data have been incorporated into the geologic framework model and the integrated site model. Data have also been provided to repository design.

Remaining work, identified in Section 3.1.1, will focus on updating and maintaining the integrated site model and providing any additional site information needed to design a repository at Yucca Mountain. This work is expected to include mapping new excavations, testing samples from new excavations and boreholes, providing support to other on-going activities and addressing specific issues that arise, monitoring, and incorporation of new data into geologic databases and models.

Tectonics and Crustal Conditions Subissue Status: What are the viable tectonic models and crustal conditions at Yucca Mountain? This subissue was partially addressed in Revision 0 resolution of the issue resolution status report on structural deformation and seismicity. NRC staff concluded that resolution has been achieved concerning the viable tectonic models for Yucca Mountain. In addition the staff concluded that methods exist to bound the range of future changes to the seismotectonic component of the geologic setting and the resulting consequences. Adequate methods are currently available to reasonably estimate future site behavior with regard to faulting, seismicity, and fracturing; geomechanical disruption of waste packages; structural controls on flow of water, moisture, heat, and magma; and stability of the geologic repository operations area and the engineered and natural barrier systems.

Most of the data related to these topics have been collected. Studies of regional tectonics and local crustal stress conditions have provided information to develop models of regional and local crustal stress. Ongoing seismic and geomechanical monitoring is expected provide data to refine interpretations and conceptual models of regional and local crustal stress.

Site characterization work planned to address structural deformation and seismicity is identified in Section 3.1.1. Subsurface design activities related to seismicity and performance confirmation (long-term drift stability) are identified in Section 3.2.1, and waste package design activities related to seismicity are identified in Section 3.2.2. Performance assessment work related to seismicity is identified in the description of disruptive events in Section 3.3.1.

4.3.3.3 Summary of the Evolution of the Near-Field Environment Key Technical Issue

The near field is considered to be the portion of the site for which changes in the physical and chemical properties, resulting from construction of the underground facility or from heat generated by the emplaced radioactive waste, affects performance of the repository (NRC 1997b). Evolution of the near field is expected to feature coupling of thermal, hydrologic, and chemical processes that could affect large portions of the mountain. The coupled processes could affect various components of the engineered barrier system and thereby affect repository system performance. Additional perturbations to the near field are expected to result from the introduction of repository construction materials and waste package materials. The key technical issue comprises four subissues identified in the section that follows.

Objective of the Key Technical Issue. The objective of the evolution of the near-field environment key technical issue is to assess all aspects of the evolution of the near-field geochemical environment that have the potential to impact the performance of the proposed repository.

Resolution of the following four subissues must address four aspects of coupled processes:

- Identify the potential effects of coupled processes on performance.
- Determine how the natural system will influence, and be influenced by, coupled processes.
- Evaluate how the engineered materials will influence, and be influenced by, coupled processes.
- Determine the adequacy of any representation of the effects of coupled processes in performance assessment.

Effects of Coupled Processes on the Rate of Seepage into the Repository Subissue Status. The description in the issue resolution status report indicates that the first and second aspects listed in the preceding paragraph, with the exception of the potential impact of microbial processes on the natural system, have been addressed, as related to this subissue. However, available information is not sufficient to address microbial activity or the third and fourth aspects as identified above. Hence, insufficient information is currently available to resolve the subissue at the NRC staff level.

In situ measurements have provided useful information about the nature of seepage, and modeling studies calibrated to these measurements have provided indications of the fraction of percolation that would actually contribute to seepage under ambient or pre-waste emplacement conditions. Estimates of seepage have considered spatial and temporal variations in percolation flux and properties of the host rock, which may change because of thermal effects. Understanding of coupled process effects on seepage appears to be adequate for the current phase of design and performance assessment, but additional work will provide information to refine seepage estimates.

DOE has work planned to predict the conditions under which water will seep into the drifts holding the waste packages as identified in Sections 3.1.2 and 3.1.5.

Effects of Coupled Processes on Waste Package Lifetime Subissue Status. The description in the issue resolution status report indicates that the first two listed aspects, with the exception of the potential impact of microbial processes on the natural system, have been addressed, as related to this subissue. However, available information is not sufficient to address microbial activity on the natural system, or the last two listed aspects. Hence, insufficient information is currently available to resolve the subissue at NRC staff level.

Evaluations of coupled processes generally have been done in the context of principal factors affecting postclosure performance or potential design options and alternatives. A common theme in both contexts is waste package lifetime as influenced by the amount of water contacting waste packages and by chemistry of water on waste packages. Limits on the amount of water contacting waste packages and chemistry of water on the waste package are discussed in Section 2.2.4.2. Based on current information, it appears that the level of understanding of coupled processes on waste package lifetime is adequate for the current phase of design and performance assessment. However, additional work is planned to support advanced stages of design and performance assessment for the LA.

DOE has work planned to address the effects of microbial activity on waste package lifetime. This work is identified in Sections 3.1.5 and 3.2.2.

Effects of Coupled Processes on the Rate of Release of Radionuclides from Breached Waste Packages Subissue Status. The description in the issue resolution status report indicates that the first two listed aspects, with the exception of the potential impact of microbial processes on the natural system, have been addressed, as related to this subissue. However, available information is not sufficient to address microbial activity or the last two listed aspects. Hence, insufficient information is currently available to resolve the subissue at the NRC staff level.

Various factors related to evaluation of the rate of release of radionuclides from breached waste packages are discussed in Section 2.2.4.2. Neptunium solubilities considered in performance assessments appear to capture the range of conditions likely at the site. DOE believes that current estimates of seepage into the waste package, integrity of fuel cladding, and in the range of possible chemistry that controls dissolution of uranium oxide and glass waste forms are adequate for current phases of design and performance assessment. However, additional planned work will improve the understanding of neptunium solubility, the formation of radionuclide-bearing colloids, and the ability to represent these effects.

DOE has work planned to investigate the effects of microbial activity on the natural system and evaluate the effects of coupled processes on radionuclide mobilization. This work is identified in Sections 3.1.5 and 3.2.2.

Effects of Coupled Processes on Radionuclide Transport through Engineered and Natural Barriers Subissue Status. The description in the issue resolution status report indicates that the first two listed aspects, with the exception of the potential impact of microbial processes on the natural system, have been addressed, as related to this subissue. However, available information is not sufficient to address microbial activity or the last two listed aspects. Hence, insufficient information is currently available to resolve the subissue at the NRC staff level.

Based on the information summarized in Section 2.2.3, mobilities of most radionuclides in the waste are low at Yucca Mountain, hence, peak doses of these radionuclides downgradient from a repository at Yucca Mountain are expected to be negligible under any conditions. A small fraction (less than 0.04 percent) of the radioactivity is mobile; however, there are a few remaining issues regarding the characteristics of the system affecting radionuclide mobility. DOE believes that representations of mobilities of radionuclides are adequate for the current phase of design and performance assessment. Additional work will strengthen design recommendations and performance analyses needed to support site recommendation and the LA.

Planned studies to model the movement of radionuclides through materials of the engineered barrier system are identified in Sections 3.1.5.

4.3.3.4 Summary of Container Life and Source Term Key Technical Issue

The primary issue is the adequacy of the design of the engineered barrier system to provide reasonable assurance that containers will be sufficiently long lived, and radionuclide releases from the engineered barrier system will be adequately controlled. The key technical issue is comprised of four subissues identified in this section.

Objective of Key Technical Issue. The objective of the key technical issue is to ensure that container design and packaging of spent nuclear fuel and other high-level radioactive waste forms are sufficient to make significant contributions to overall repository performance. This issue resolution status report addresses a component of the effects of corrosion subissue.

What are the Effects of Corrosion on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment? This subissue considers container failure by various corrosion modes and comprises seven components:

- Dry oxidation of container materials
- Humid air corrosion and aqueous corrosion, including general corrosion
- Localized corrosion
- Stress corrosion cracking
- Galvanic corrosion

- Microbial corrosion

- Hydrogen embrittlement

The issue resolution status report addressed only the significance of dry oxidation of container materials during the dry period of the proposed repository. In the issue resolution status report (NRC 1998a), NRC staff concluded that dry oxidation is not a significant failure mode or degradation process for container materials. The balance of the components of this subissue will be addressed in later revisions of the issue resolution status report.

Studies of the near-field geohydrology and the humidity and temperature at the waste package have provided estimates of the amount of water that could drip onto waste packages and estimates of the range of humidity conditions expected at the waste package (see Section 2.2.4.2). Additional information is needed to increase the realism in the model prior to site recommendation and license application.

Work planned by DOE and related to this subissue is focused on studies of accelerated corrosion in humid conditions (less than 100 percent humidity and a constant temperature). The relative humidity at which the transition occurs from dry oxidation to aqueous film electrochemical corrosion will be determined. This work is identified in Section 3.2.2.9.

What are the Effects of Materials Stability and Mechanical Failure on the Lifetime of the Containers and the Release of Radionuclides to the Near-Field Environment? This subissue considers long-term degradation of material properties of containers subjected to elevated temperatures for a prolonged period. Such degradation may include redistribution and segregation of impurities and may result in thermal embrittlement leading to early mechanical failure by brittle failure. The components of this subissue will be addressed in later revisions of the issue resolution status report.

DOE believes that information about the range of chemistry of water on the waste package and on estimates of the integrity of the waste package barriers is adequate to support the current phase of design and performance assessment. However, additional work is planned to support development of the design basis for the LA.

Work planned by DOE to address this subissue is identified in Section 3.2.2.9.

Is Spent Nuclear Fuel Sufficiently Resistant to Degradation to Contribute to the Control of Radionuclide Releases to the Near-Field Environment? This subissue considers degradation of spent nuclear fuel and subsequent radionuclide release from spent nuclear fuel, in dry air and aqueous environments following failure of fuel cladding. Release mechanisms considered include matrix dissolution, solubility limits, colloid formation, dry oxidation of spent nuclear fuel, and gaseous diffusion. The components of this subissue will be addressed in later revisions of the issue resolution status report.

Various factors related to integrity of spent fuel cladding are discussed in Section 2.2.4.2. Because of dose sensitivity to cladding integrity, DOE is concerned about whether the current model is realistic or spans the range of possibilities. Although significant uncertainties remain, estimates of seepage into the waste package, integrity of fuel cladding, and the range of possible chemistry that controls dissolution of uranium oxide and glass waste forms appear to be adequate for the current phase of design and performance assessment. Current representations of radionuclide-bearing colloids and transport of radionuclides within and outside waste packages could be improved by additional work.

Work planned by DOE to address this issue is identified in Section 3.2.2 under Waste Form Testing and Modeling.

Is High-Level Radioactive Waste Glass Sufficiently Resistant to Degradation to Contribute to the Control of Radionuclide Releases to the Near-Field Environment? This subissue considers degradation of high-level radioactive waste glass and includes release mechanisms of matrix dissolution, solubility limits, and colloid formation. The components of this subissue will be addressed in later revisions of the issue resolution status report.

As discussed in the previous subissue, DOE believes that estimates of the range of possible chemical controls on dissolution of uranium oxide and glass waste forms are adequate for the current phase of design and performance assessment. Representations of radionuclide-bearing colloids for design and performance assessment could be improved by additional work.

Work planned by DOE to address this issue is identified in Section 3.2.2.9.

4.3.3.5 Summary of Thermal Effects on Flow Key Technical Issue

The thermal effects on flow key technical issue is considered to be an important factor in evaluating repository performance (NRC 1997d). The key technical issues focuses on two potential concerns for performance assessment:

- Thermally-driven redistribution of moisture through partially saturated, fractured, porous media caused by the emplacement of heat-generating high-level radioactive waste
- Temperature and humidity of the waste package environment

This key technical issue comprises three subissues identified in this section.

Objective of the Key Technical Issue. Generally, the objective of the key technical issue is to evaluate the potential for either extended periods of dryness in the proposed repository or for channeling of moisture toward emplacement drifts during either the period of high heat or during cooling. To evaluate this potential, it is necessary to understand the spatial and temporal effects of the thermal loading on liquid and gas phase flux, and resultant effects on temperature and humidity of the waste package environment.

Is the DOE Thermal-Hydrologic Testing Program, Including Performance Confirmation Testing, Sufficient to Assess the Potential for Thermal Reflux to Occur in the Near Field? This subissue remains open, but NRC staff has identified no major concerns with the design of DOE thermal-hydrologic tests. However, NRC staff indicated the need for qualification of all information and analyses necessary to evaluate the effect of thermally-perturbed fluid flux on repository performance. NRC activities related to this subissue are expected to be limited to monitoring of the progress of the thermal-hydrologic tests.

DOE performance assessment sensitivity studies indicate that thermal effects on moisture movement are likely to occur when the waste packages are intact. If correct, this interpretation suggests that these effects would be expected to have minimal effects during the postclosure performance period. However, longer-term effects on the natural barriers are under study to ensure that performance assessment considers the appropriate range of conditions.

Planned DOE testing related to thermal effects on flow is identified in Section 3.1.4. Model abstraction work related to thermohydrology is identified in Section 3.3.1.

Is the DOE Thermal-Hydrologic Modeling Approach Sufficient to Predict the Nature and Bounds of Thermal Effects on Flow in the Near Field? This subissue remains open, and NRC staff is expected to focus its activities related to this key technical issue on the modeling approach subissue.

DOE believes that the current understanding of thermal-hydrologic effects is adequate to support the current phase of design and performance assessment. Studies indicate that thermal effects on moisture redistribution are mitigated by robust waste packages that are designed to withstand these effects and remain intact during the postclosure performance period. However, longer-term effects on the natural barriers are under study to ensure that performance assessment considers the appropriate range of conditions. In situ testing (single heater test) has shown that this testing, coupled with calibrated modeling, provides a useful tool for determining the effects of heat on the moisture content in the immediate vicinity of the heater. Additional testing in the drift scale test is expected to confirm the validity of this approach. Predictions of moisture redistribution are being compared to actual effects, including the movement of water away from the heaters and the collection of water in test boreholes. Continued testing and modeling are expected to provide substantial information about the potential for dryout of the rock around the emplacement drifts and increase the confidence in the representation of these effects for performance assessment.

Planned DOE work related to modeling thermal effects on flow is identified in Section 3.1.4. Model abstraction work related to incorporation of the process model or models into TSPA is identified in Section 3.3.1.

Does the DOE Total System Performance Assessment Adequately Account for Thermal Effects on Flow? This subissue remains open, and NRC staff is expected to focus its activities related to this key technical issue on the modeling approach subissue.

DOE performance assessment sensitivity studies indicate that thermal effects on moisture are likely to occur when the waste packages are intact. If correct, this interpretation suggests that thermal effects on moisture movement would be expected to have minimal effects during the postclosure performance period. However, longer-term effects on the natural barriers are under study to ensure that performance assessment considers the appropriate range of conditions. Work from the drift scale test, from modeling, and from natural analog studies is expected to increase the level of confidence in the representation of effects of heat on the flow system.

Model abstraction work planned by DOE to address thermal hydrology, evolution of the near-field environment, and waste package degradation is identified in Section 3.3.1. Work planned to incorporate the model abstractions into TSPA is also identified in Section 3.3.1.

4.3.3.6 Summary of Repository Design and Thermal-Mechanical Effects Key Technical Issue

The time-dependent thermal-mechanical coupled response of a jointed rock mass is a significant consideration for repository design and postclosure performance assessment. Repository design for adequate postclosure performance requires an understanding of the thermal-mechanical response of the jointed rock mass over a period of hundreds to thousands of years. The long-term thermal-mechanical response is expected to influence evolution of the near-field environment, waste package life, and mobilization and transport of radionuclides. Design for preclosure operations requires an understanding of the thermal-mechanical response of the jointed rock mass as it influences drift, shaft, and ramp stability, and waste retrievability. This key technical issue comprises four subissues identified in this section.

Objectives of the Key Technical Issue. The objectives of this key technical issue are captured in the subissue status statements in this section.

Implementation of an Effective Design Control Process within the Overall Quality Assurance Program Subissue Status. The issue resolution status report for this key technical issue addressed the resolution status of one component of this subissue, namely the design control process for the Exploratory Studies Facility (NRC 1997e). NRC staff concluded that the DOE design control process for the Exploratory Studies Facility is acceptable.

The design control process has been the subject of several interactions between DOE and NRC. DOE believes the design control process is adequate to support the development of the design for the LA.

Planned DOE work related to design of subsurface facilities is identified in Section 3.2.1.

Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption (Including Implications for Drift Stability, Key Aspects of Emplacement

Configuration, such as Fault Offset, Distance, Retrievability, and Waste Package Damage). Three components have been identified by NRC staff for this subissue:

- The DOE method to assess seismic and fault displacement hazard
- The DOE seismic design method
- Seismic and fault displacement inputs to design and performance assessment

This issue resolution status report addressed part of the component related to the adequacy of the seismic design method. NRC staff concluded that the DOE preclosure seismic design method is acceptable contingent on review of the final topical report related to the seismic design method. The design inputs component is being addressed by NRC staff in the structural deformation and seismicity key technical issue.

DOE seismic design standards have been formalized based on input from the seismic design working group. The standards have been incorporated into the repository design process. Studies indicate that seismic effects on performance would be minor. Disruptive events related to seismicity are described in Section 2.2.1.3.

Planned DOE work related to design of subsurface facilities is identified in Section 3.2.1.

Consideration of Thermal-Mechanical Effects on Underground Facility Design and Performance (Including Implications for Drift Stability, Key Aspects of Emplacement Configuration That May Influence Thermal Loadings and Associated Thermal-Mechanical Effects, Retrievability, and Changes in Hydrologic Properties That May Influence the Near-Field Environment). This subissue comprises two components:

- Stability of the underground excavations with regard to safety during the preclosure period, waste retrievability, and potential adverse effects on emplaced wastes
- Changes in hydrological properties of fractures caused by thermal-mechanical perturbation of the rock mass that might adversely affect the near-field environment

NRC will address the first component under the repository design and thermal-mechanical effects key technical issue. The second component will be addressed under the thermal effects on flow key technical issue.

DOE is testing ground control materials for use in the design for the LA. Testing of concretes for mechanical and chemical properties is being done under a variety of thermal conditions that alter the hydrologic properties of the concretes and affect the ground support properties.

Planned DOE work related to design of subsurface facilities is identified in Section 3.2.1. Planned DOE testing related to thermal effects on flow is identified in Section 3.1.5. Waste package degradation testing is identified in Section 3.2.2. Planned work related to modeling thermal effects on flow is identified in Section 3.1.5, and model abstraction work is identified Section 3.3.1.

Design and Long-Term Contribution of Repository Seals in Meeting the Postclosure Performance Objectives (Including Implications for Inflow of Water and Release of Radionuclides to the Environment). This subissue comprises three topics:

- Design and construction of seals (including material selection)
- Long-term stability of seals and their components
- Importance of seals in meeting the postclosure performance objectives

These topics will be addressed by NRC staff under the repository design and thermal-mechanical effects and the TSPA and integration key technical issues, but descriptions of which topics are to be treated under each key technical issue are not yet available.

Preliminary work on seals has been done, but seals are not viewed as crucial to the safety case for the LA. Seals are mostly to reduce the potential for human intrusion, which will be considered by TSPA in a stylized scenario. There is ongoing consideration of seals as enhancements to the defense in depth concept.

Work planned by DOE related to performance confirmation of seal performance is identified in Section 3.1.6. Seal design activities are identified in Section 3.2.1. However, specific performance assessment sensitivity study activities related to evaluation of the importance of seals in meeting the performance objective have not been identified.

4.3.3.7 Summary of Total System Performance Assessment and Integration Key Technical Issue

TSPA provides the basis for DOE demonstration of compliance with applicable standards for the disposal of high-level radioactive waste in a geologic repository. TSPAs for a geologic repository must consider the behavior of the engineered system specified in the design, important natural features of the site, combinations of disruptive events, coupling of physical processes, and possible changes to the flow and transport system. To ensure that the risk to public health and safety from a repository is fully quantified and understood, repository performance must be reflected in the modeling from a total system perspective. The integration aspect of the key technical issue ensures that information from many technical disciplines related to characterization of site features, events and processes and design of the engineered structures is appropriately included in process models and abstracted for TSPA purposes. As described in the issue resolution status report (NRC 1998c), the key technical issue comprises three subissues as described in this section.

Objective of Key Technical Issue. The objective of the TSPA and integration key technical issue is to describe an acceptable method for conducting assessments of repository performance. The focus is on those aspects of the TSPA method necessary to build an acceptable safety case and demonstrate compliance.

Model Abstraction Subissue Status. This subissue focuses on the information and technical needs related to the development of abstracted models for TSPA, in particular the need to incorporate numerous features, events, and processes into the performance assessment and the integration of those factors to provide a comprehensive analysis of the total system. Three aspects of the abstraction process are included:

- Data used to develop conceptual approaches or process-level models that are the basis for abstraction. Revision 0 of the issue resolution status report addresses the input information.

- Resulting abstracted models used to perform the TSPA. Model abstraction parts of the model abstraction subissue.

- Overall performance of the repository system as estimated in a TSPA.

The report describes two programmatic and five technical criteria related data development and model abstraction that NRC staff will use to review DOE TSPAs.

Model abstraction and sensitivity analyses have been completed for the TSPA for the VA. The TSPA for the VA represents a significant advancement over previous TSPA iterations in terms of abstraction and sensitivity analyses. Further work is needed to reevaluate the abstractions and sensitivities, and to develop revised abstractions for the site recommendation and the LA.

Work planned by DOE for the major components of TSPA is identified in Section 3.3.1. TSPA analyses tasks are identified in Section 3.3.2.

Scenario Analysis Subissue Status. This subissue considers the proper inclusion or exclusion of possible features, events, and processes that could affect repository performance, the combination of those events into scenario classes, and presentation of the results generated from various credible scenarios.

This subissue will be addressed by NRC staff in subsequent revisions of the issue resolution status report.

The first set of scenarios has been developed to support the TSPA for VA. Remaining issues will be incorporated into scenario revisions or in the development of additional scenarios, if needed, to support the TSPA for the site recommendation and the LA.

Planned DOE work related to the development and use of scenarios is identified in Sections 3.3.1 and 3.3.2.

Transparency and Traceability of the Analysis Subissue Status. This subissue emphasizes staff expectations for the contents of the DOE TSPA to support an LA. The focus of this subissue is on those aspects of the TSPA that will allow for an independent analysis of the results.

This subissue will be addressed by NRC staff in subsequent revisions of the issue resolution status report.

Efforts to improve transparency of analyses and traceability of data have been included in the TSPA for the VA. The results indicate a significant improvement over previous TSPA iterations. Additional work will include efforts to improve clarity of presentations.

Work planned by DOE that is related to transparency and traceability of analyses is identified in the descriptions of TSPA analyses in Section 3.3.2 and quality assurance for the TSPA for the LA in Section 3.3.3.

4.3.3.8 Activities Related to Development of the U.S. Environmental Protection Agency Yucca Mountain Standard Key Technical Issue

Title VIII of the Energy Policy Act of 1992 directed EPA to develop a site-specific health-based radiation protection standard for Yucca Mountain based upon and consistent with recommendations provided by the National Academy of Sciences. NRC staff is coordinating with EPA to ensure development of reasonable and implementable high-level radioactive waste standards, considering the recommendations of the National Academy of Sciences and congressional direction found in Title VIII. Once EPA issues the final standards, NRC must modify its technical requirements and criteria, as necessary to be consistent with the standards.

Based on information from NRC, DOE does not expect to receive an issue resolution status report for this key technical issue.

4.3.3.9 Summary of Unsaturated and Saturated Flow Under Isothermal Conditions Key Technical Issue

Yucca Mountain is being evaluated as a site for a geologic repository in part because of the favorable hydrogeologic conditions provided by its 700-m (2,300-ft) thick unsaturated zone. Prevailing meteorological conditions, along with local geologic conditions and plant communities, control the rates of infiltration, deep percolation, and strongly influence groundwater seepage through a geologic repository located in an unsaturated environment. As described in the issue resolution status report (NRC 1997c), the key technical issue comprises six subissues identified in this section.

Objectives of the Key Technical Issue. The objectives for this key technical issue include assessing:

- All aspects of the ambient hydrogeologic regime at Yucca Mountain that have the potential to compromise the performance of the proposed repository

- The adequacy of the DOE characterization of key site- and regional-scale hydrogeologic processes and features that may adversely affect performance.

What is the Likely Range of Future Climates at Yucca Mountain? This subissue was addressed in the Issue Resolution Status Report on Methods to Evaluate Climate Change (NRC 1997g). NRC staff concluded that methods based on paleoclimatic and paleohydrologic data can be used to bound the range of past climates in the Yucca Mountain region. NRC and DOE consider this subissue resolved.

DOE plans only limited work needed to complete process models and incorporate those models into the TSPA. Only minimal site work has been planned; the work consists of closeout activities to ensure the program is completed in a controlled fashion. Work planned to support TSPA model abstraction and related to this subissue are identified in Section 3.3.1.

What are the Likely Hydrologic Effects of Climate Change? This subissue was addressed in the Issue Resolution Status Report on Methods to Evaluate Climate Change (NRC 1997g) and summarized in the Issue Resolution Status Report on Unsaturated and Saturated Flow Under Isothermal Conditions (NRC 1997c). NRC staff concluded that methods based on paleoclimatic and paleohydrologic data can be used to bound the range of likely hydrologic effects of climate change. DOE and NRC consider this subissue resolved.

DOE plans only limited work needed to complete process models and incorporate those models into the TSPA. Site characterization work planned by DOE and related to this subissue is identified in Sections 3.1.1, 3.1.2 and 3.1.3. Work planned to support TSPA model abstraction and related to this subissue are identified in Section 3.3.1.

What is the Estimated Amount and What is the Spatial Distribution of Present-Day Shallow Groundwater Infiltration? This issue was addressed in the Issue Resolution Status Report on Unsaturated and Saturated Zone Flow Under Isothermal Conditions (NRC 1997c). NRC staff concluded that various methods are available to bound the range of present-day infiltration at Yucca Mountain, and the methods are basically the same as those used by DOE to evaluate shallow infiltration. DOE and NRC consider this subissue resolved.

DOE believes that the appropriate range of infiltration has been captured for the current phase of design performance assessment.

DOE plans only limited work needed to complete process models and incorporate those models into the TSPA. Site characterization work planned by DOE and related to this subissue is identified in Sections 3.1.1 and 3.1.2. Work planned to support TSPA model abstraction and related to this subissue are identified in Section 3.3.1.

What is the Estimated Amount and What is the Spatial Distribution of Present-Day Groundwater Percolation Through the Proposed Repository Horizon? This subissue is scheduled to be addressed by NRC staff in the next revision of the Issue Resolution Status Report on Unsaturated and Saturated Zone Flow Under Isothermal Conditions (NRC 1997c).

A discussion of the status of information about percolation is presented in Section 2.2.4.2. DOE believes that the current understanding of average percolation flux, and estimates of flow are adequate for the current phase of design and performance assessment. Additional work is planned to improve estimates that support site recommendation and the LA.

DOE plans to address this subissue by completing development of the unsaturated zone model and abstraction of that model into TSPA. Site characterization work planned by DOE and related to this subissue is identified in Sections 3.1.1 and 3.1.2. Work planned to support TSPA model abstraction and related to this subissue are identified in Section 3.3.1.

What is the Estimated Amount and What is the Spatial Distribution of Groundwater Percolation Through the Proposed Repository Horizon During the Period of Repository Performance? This subissue is scheduled to be addressed by NRC staff in the next revision of the Issue Resolution Status Report on Unsaturated and Saturated Zone Flow Under Isothermal Conditions (NRC 1997c).

A discussion of information about percolation of water to depth at Yucca Mountain is presented in Section 2.2.4.2. DOE believes that the current understanding of average percolation flux, and estimates of flow are adequate for the current phase of design and performance assessment. Additional work is planned to improve estimates that support site recommendation and the LA.

DOE plans to address this subissue by testing, including thermal testing and investigation of thermal-hydrologic coupling and studies of the near-field environment, development of a thermohydrology model, and abstraction of the thermohydrology model for TSPA. Site characterization work planned by DOE and related to this subissue is identified in Sections 3.1.1, 3.1.2 and 3.1.5. Work planned to support TSPA model abstraction and related to this subissue is identified in Section 3.3.1.

What are the Ambient Flow Conditions in the Saturated Zone? This subissue is scheduled to be addressed by NRC staff in the next revision of the Issue Resolution Status Report on Unsaturated and Saturated Zone Flow Under Isothermal Conditions (NRC 1997c).

Current information indicates that flow from the location of the potential repository is generally to the southeast, but additional information is needed to corroborate interpretations about flow paths and volumes, mixing of waters and other saturated zone factors.

DOE plans to address this subissue by completing development of the saturated zone flow and transport model and by abstraction of the model for TSPA. Site characterization work planned by DOE and related to this subissue is identified in Sections 3.1.1 and 3.1.3. Work planned by to support TSPA and related to this subissue is identified in Section 3.3.1.

4.3.3.10 Summary of Radionuclide Transport Key Technical Issue

The possibility of radionuclide transport through the subsurface from the repository to the accessible environment is a fundamental concern in evaluating Yucca Mountain as the potential site for a high-level radioactive waste repository. Dose calculations like those recommended by the National Academy of Sciences require estimates of concentrations of different radionuclides in aqueous, gas, and solid phases that could occur at different times along exposure pathways. The main controls on radionuclide transport are system geochemistry and hydrology, and therefore, processes that control the concentrations of radionuclides in groundwater must be evaluated in any assessment of repository performance. The key technical issue comprises four subissues as identified in the following section. Note that DOE has not received the issue resolution status report for this key technical issue. The information about the key technical issue and associated subissues has been extracted from the NRC annual report for fiscal year 1996 (NRC 1997a).

Objective of Key Technical Issue. Work under this key technical issue is intended to lead to resolution of the associated subissues through the use of quantitative models to evaluate the effects of site-specific transport and dilution processes on repository performance.

Identifying Which Radionuclides Require Some Form of Retardation to Meet Performance Requirements at Yucca Mountain. DOE has not yet received the issue resolution status report for this key technical issue. DOE expects this subissue to be revised when the issue resolution status report is issued.

Based on the information summarized in Section 2.2.3, mobilities of most radionuclides are low at Yucca Mountain, and peak doses of these radionuclides downgradient from a repository at Yucca Mountain are expected to be negligible under any conditions. Exceptions include two soluble fission products, technetium-99 and iodine-129, which do not readily interact with the rock, an actinide, neptunium-237, that sorbs weakly on minerals at Yucca Mountain, and two radionuclides, plutonium-239 and plutonium-242, that attach to colloidal particles which travel readily with water. Presence of technetium-99 and iodine-129 depends on degradation of the waste form. DOE believes that adequate information exists about the degradation of cladding and estimates of neptunium-237 solubility to support the current phase of design and performance assessment. However, additional planned work will improve the understanding of neptunium solubility, the formation of radionuclide-bearing colloids, and the ability to represent these effects.

Site and design work planned to address radionuclide transport is identified in Sections 3.1.5 and 3.2.1.3. Waste package design work addressing radionuclide transport is identified in Section 3.2.2. TSPA work planned to address radionuclide transport in the unsaturated and saturated zones is identified in Sections 3.3.1 and 3.3.2.

Evaluating Geochemical and Hydrological Controls on Radionuclide Transport and the Potential for Dilution at Yucca Mountain. DOE has not yet received the issue resolution status report for this key technical issue. DOE expects this subissue to be revised when the issue resolution status report is issued. Representations of the biosphere and dilution of radionuclides resulting from pumping are adequate for the current phase of design and performance assessment. Additional laboratory testing regarding non-reversible sorption to colloids and filtration effects, supplemented by computer modeling could significantly improve understanding of the ability to represent formation and transport of colloids in the unsaturated zone. A field tracer test at Busted Butte, near the site, is underway to examine the transport characteristics of unsaturated rocks similar to those underlying the repository. These tests examine both reactive and conservative tracers analogous to key radionuclides in the repository system. Microsphere tests examine transport properties of colloids. The tests are expected to improve confidence in the representation of advective and diffusive transport effects in the unsaturated zone.

Work planned by DOE to address unsaturated zone geochemical and hydrological controls on radionuclide transport is identified in Section 3.1.2.

Evaluating Conceptual Models and Mathematical Approaches to Modeling Radionuclide Retardation at Yucca Mountain. DOE has not yet received the issue resolution status report for this key technical issue. DOE expects this subissue to be revised when the issue resolution status report is issued.

DOE is currently using a linear, infinite capacity sorption model with a minimum sorption distribution coefficient (K_d). DOE believes that this model is adequate for the TSPA for the VA and expects to refine parameter selection for the model to support the TSPA for the LA. DOE expects this subissue to be revised when the issue resolution status report is issued.

Planned site characterization related to developing conceptual models of radionuclide retardation in the unsaturated zone is identified in Section 3.1.2. TSPA work related to this subissue is identified in Sections 3.3.1 and 3.3.2.

Evaluating the Sensitivity of the Overall Performance of the Potential Repository to Ranges in Parameters Controlling Radionuclide Transport. DOE has not yet received the issue resolution status report for this key technical issue. DOE expects this subissue to be revised when the issue resolution status report is issued.

In the TSPA-VA, DOE has evaluated the sensitivity of overall performance of the potential repository to ranges in parameters controlling radionuclide transport. These sensitivities will be reevaluated and revised, as appropriate, to support the site recommendation and the LA.

Work planned by DOE to address repository performance sensitivity to parameters affecting radionuclide transport is identified in Section 3.3.2.

4.3.3.11 Communication

This section discusses the communication between NRC and DOE and how this communication will be used as DOE moves from the VA to submitting the LA to NRC. The focus of these interactions will be on early resolution of issues, at the staff level. These interactions will be conducted according to the DOE and NRC procedural agreement for communications during site characterization (DOE and NRC 1993). Meetings will be open for the public to observe, consistent with NRC policy on open meetings. As discussed in the procedural agreement, NRC will provide timely notification of meetings to the host state, affected units of local government, and affected Indian tribes. A written report agreed to by DOE and NRC will be prepared for each formal interaction.

Before the LA is submitted, joint meetings will focus on the following:

- Reaching agreement on approaches to technical methods and level of detail to be used for selected aspects of the safety assessment

- Developing a common understanding between DOE and NRC on the content of the LA

DOE will give NRC information in a timely manner through formal and informal meetings. DOE will continue to work with NRC to reach early resolution of technical issues before submitting the LA. Regularly scheduled joint meetings will continue to be conducted. Additional meetings with NRC will be scheduled and conducted, as needed, to facilitate timely NRC review of DOE information and resolution of issues.

DOE expects to continue quarterly management and technical meetings with NRC between now and submission of the LA. The management meetings provide a forum for discussion of licensing, management issues, and concerns. The quarterly technical meetings provide a forum for discussion of site characterization, performance assessment, repository design, and scientific studies.

In addition to the quarterly meetings, DOE will hold technical exchanges and various other less formal meetings. These interactions with NRC will facilitate timely NRC review of key project information and support early resolution of issues. DOE and NRC jointly agree semiannually to the timing and focus of these meetings. Also, DOE staff expects to continue meeting informally with NRC onsite representatives. After the LA has been submitted, communications will focus on accomplishing the DOE portion of the NRC review. This will include formal submittal by DOE to NRC of additional information in the form of LA amendments, in response to NRC requests for additional information and other needs to supplement the LA. This will also include informal meetings, open to the public, to facilitate the NRC review.

Table 4-2 identifies sections of the VA in which discussions of aspects of the key technical issues are found. The table serves two purposes: it acts as a crosswalk between the issues and discussions in various parts of the VA, and it shows how VA integrates discussions relevant to each issue.

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5. SUPPORT ACTIVITIES

Major activities that support submittal of the LA, in addition to those discussed in previous sections, are discussed in this section. The schedule for these activities is in Section 7, and the cost for these activities is in Section 6.

5.1 FIELD CONSTRUCTION AND OPERATIONS ACTIVITIES

This section summarizes the field construction and operations activities needed between the VA and submission of an LA. The DOE plans identify the specific construction and operations activities required to prepare for LA submittal.

At the same time, construction and operations activities in the field will continue to support ongoing site characterization. To meet these objectives, DOE has established a field infrastructure that must be maintained in a manner consistent both with the principles of sound stewardship of federal property and with DOE site characterization goals. The overall facility infrastructure involves three general areas:

- A controlled land area, with a number of buildings and other facilities in use, and the general infrastructure, such as roads, utilities, and communication systems in, and adjacent to, Area 25 of the Nevada Test Site
- Surface test facilities, including boreholes and test excavations
- The Exploratory Studies Facility, surface support facilities, utilities, and underground openings created for subsurface site characterization

The construction and operations field staff supports ongoing scientific activities, maintains site infrastructure, and provides operations and maintenance support services. The staff also maintains limited services that could handle future construction. A detailed description of the general areas and facilities follows. Also included are summaries of the expected activities, services, and staff requirements necessary to operate, maintain, and protect the systems, structures, and components associated with each area.

5.1.1 Land Area, Facilities, Buildings, and Infrastructure

For scientific investigation, DOE controls a 90-mile^2 portion of Area 25 of the Nevada Test Site, some small blocks of the U.S. Air Force range, and a portion of BLM property. These three property owners allow access and use of this land through memoranda of understanding and rights-of-way agreements.

The facilities at the central support area were constructed in the 1960s for the Nuclear Rocket Development Program. This complex contains 16 buildings and associated facilities supporting numerous DOE functions, such as sample management, laboratory testing, and field administrative control. Storage yards are maintained for drilling materials and ancillary equipment. Utilities include power lines and substations, water lines and storage tanks, sewer lines, sewage lagoons, and two water wells used primarily by DOE. Field and remote radio communications, and some phone and data transfers, are supported by automatic repeater stations at strategic, elevated locations within the controlled area. The transportation network consists of 20 miles of paved roads and 28 miles of unpaved roads. Bechtel Nevada, the support contractor for the Nevada Test Site, also provides requested services to maintain these facilities. In the context of a potential repository, the facilities are considered temporary because many of their elements do not meet current technical codes and standards.

5.1.2 Surface Testing Facilities

One important type of surface testing facility is a drill pad and associated equipment for the drilling and monitoring of a borehole. Boreholes provide ongoing subsurface geologic and hydrologic access or contain various testing instrumentation. Other types of surface testing facilities include trenches, pits, environmental plots, and geologic exposures. There are also environmental plots at several locations within the controlled area to test DOE's ability to stabilize and reclaim disturbed areas.

Trenches and pits were excavated for surface investigations of recent geologic faulting, climatology, surface hydrology, and mineralization. Several acres of geologic exposures, cleaned of all surface material, are being used to evaluate faulting and structure in the rock.

An access control program protects the boreholes and limits access to them. A site maintenance program maintains safety around pits and trenches before they are backfilled and the area reclaimed.

5.1.3 Exploratory Studies Facility

The Exploratory Studies Facility portal and surface facilities are located within Area 25 of the Nevada Test Site. The underground facilities extend westward under the Yucca Mountain crest into property formerly controlled by the Air Force and the BLM. The above-ground facilities, located on a multiacre pad, include surface buildings and temporary structures near the North Portal, a concrete batch plant, a casting yard, a gravel pit, muck storage areas, a water delivery system, and a sanitary sewage system. Also included are the underground openings to the Exploratory Studies Facility itself.

Several essential services, facilities, and personnel are found on the pad. These include DOE management, contractors, craft laborers, quality control, testing organizations, underground instrument assembly and setup, calibration support, surveying, maintenance and repair, underground utility feeds, medical and emergency response, engineering, warehousing, staging for underground activities, and change-house facilities.

The primary underground feature of the Exploratory Studies Facility is a 7.9-km (5-mile) long, 7.6-m (25-ft) diameter tunnel, including ramps and the main drift. A 2.8-km (1.7-mile) repository block cross drift and various alcoves used for testing are being excavated from the primary 7.9-km (5-mile) loop. Support equipment in the underground facility includes a ventilation system, a power-distribution system, water lines, pumping-discharge lines, compressed-air lines, haulage track, and lighting. Other equipment serves the need for data acquisition, fire protection, and communications.

After the completion of the tunnel boring machine operations in the cross drift, the following work on the Exploratory Studies Facility will take place:

- Constructing the cross drift alcoves and test areas
- Completing configuration of all unfinished systems
- Completing drawings of the final Exploratory Studies Facility
- Developing final operations and maintenance-related information

5.1.4 General Services and Staffing

Buildings and facilities at the site are routinely assessed to assure proper and safe operation. Periodic assessments are also performed in the following areas:

- Planning and scheduling of maintenance
- Issuing work instructions
- Tracking work progress and completion
- Overseeing work in progress
- Performing major repairs
- Planning overhaul or replacement of aging systems

Personnel must be familiar with the site facilities, capabilities, and needs. They must also work with the Nevada Test Site on mutually needed support services, including the following:

- Safeguards and security
- Communications
- Housekeeping, sanitation, vermin control

- Radiological clearance and control
- Environmental assessments
- Gas and oil delivery
- Transportation
- Emergency response and incident reporting
- Materials and property control
- Training
- Information security program

5.1.5 Services for Surface Testing Facilities

The controlled-area operations and maintenance staff are responsible for approximately 500 testing facility sites, including boreholes, trenches, and pits. Operations and maintenance for surface testing facilities include structures, systems, equipment, boreholes, pads, pits, trenches, and geologic exposures. The staff must comply with safety, health, environmental, test interference, security, and waste containment requirements to ensure the quality of data intended for use in an LA. Operations and maintenance staff are also responsible for the following:

- Controlling borehole access in accordance with quality assurance requirements

- Refitting and remediating historic, active, and inactive surface testing facilities to conform with protection requirements

- Coordinating use of field testing facilities for participants

- Performing quality control evaluations of testing facility locations, as well as access to them

- Maintaining historical records of the locations, dimensions, and chronology of construction, maintenance, and testing activities at surface testing facilities in accordance with quality assurance requirements

5.1.6 Services for the Exploratory Studies Facility

Services for the Exploratory Studies Facility include the following:

- Operating the facility and providing general housekeeping

- Controlling access

- Promoting safety and health

- Operating underground haulage, transportation, and ventilation

- Maintaining ground support and control in accordance with quality assurance requirements

- Providing equipment repair, drilling and testing support, and utility maintenance and modification

- Providing a capability for limited construction apart from major excavation work

- Maintaining an underground infrastructure, equipment, and materials for future activities in the Exploratory Studies Facility

- Providing mine rescue and safety equipment

- Providing materials testing and calibration services

- Providing technical overview and support

- Integrating site characterization testing activities with operations of the Exploratory Studies Facility

5.2 OTHER SUPPORT ACTIVITIES

In addition to field construction and operations, other activities necessary to support YMP include information technology, program information management, quality assurance, project management, institutional interactions, administrative services, and training support.

5.2.1 Information Technology

The information technology staff maintains the DOE information base and makes it readily available through the computing and telecommunications infrastructure. In addition, the information technology staff supports DOE through computer operations and electronic databases by providing and maintaining the reliable and robust computing and telecommunications systems that will be essential to the implementation of the plans for the activities needed to support site recommendation and the LA.

- System administration, data management, and software development and maintenance are provided to support the requirements of DOE in accordance with requirements set forth by federal regulations, DOE orders, and Circular A-130 (Office of Management and Budget 1996).

- The information technology compliance function provides a framework to account for all information technology investments and manage them as capital assets.

- The enterprise and special purpose server operations function establishes and maintains a server-based, enterprise-wide information architecture to support the production of complex products such as the LA. The servers are the hardware foundation allowing the enterprise-wide systems necessary to comply with the DOE information architecture baseline. These servers are necessary to manage information required to produce complex products and documents such as the LA. Management and maintenance of these servers and the associated data dissemination function are critical to the success of this program.

- Network and telecommunications operations is responsible for the design, implementation, and operation of the telecommunications infrastructure, providing for information dissemination and interoperability throughout DOE. This function includes support for data networks, video systems, voice communications, public access, and security. These activities support the proposed revision to the NRC regulation (10 CFR, Part 2, Subpart J) requiring DOE to develop a telecommunications network in support of the LA.

- Technology assessment and migration is conducted to ensure that the information architecture continues to meet the needs of DOE and its users. This includes the assessment and evaluation of new technologies to address new and existing requirements, taking into consideration compatibility and interoperability with the existing information architecture baseline.

- Conducting business management oversight program reviews in key areas such as information systems, records management, telecommunications, computer security, and operations ensures that functions are being performed efficiently and effectively and that there are continuous efforts towards process improvement.

- Hotline, desktop and technical services are provided to maintain an effective, efficient, and functional computing environment. This support complies with DOE policy (Order 200.1) and Circular A-130 (Office of Management and Budget 1996) that require support for the operating elements of this program in accomplishing mission-essential tasks in an effective and efficient manner. This function is indispensable in a program of this magnitude.

- The planning, compliance, and information security function encompasses information resources planning, performance-based compliance assessments, and information security requirements as mandated by DOE (Orders 200.1, 224.1, 470.1, Safeguards and Security Program, 471.2A, Information Security Program, and 1360.2B) and Circular A-130 (Office of Management and Budget 1996). A primary objective of information resources planning is to ensure that information resources are acquired, implemented, and available to support project requirements when needed.

The ongoing objectives of information management include providing a reliable computing environment; ensuring the integrity of data and systems; and providing for enterprise-wide interoperability and data dissemination via a reliable telecommunications infrastructure, and the adoption of hardware and software standards. Beyond these daily objectives, the most important long-term goal is to develop a licensing support network that will provide the distributed data-dissemination instrument envisioned by NRC in the proposed revision to 10 CFR 2, Subpart J.

5.2.2 Program Information Management

Preparing a well-documented site recommendation and LA requires an integrated and controlled system for managing information in many forms, including data, reports, records, computer programs, and reference documents. This system is provided through the use of procedures meeting OCRWM quality assurance requirements.

To this end, document management, program records, configuration management, technical data management, and a technical information center are maintained. These systems are described as follows:

- Document management provides production services related to the life cycle of documents. This includes document control, publications, graphics, reprographics, technical information center, and web publishing services. Activities include directing the formal review and approval processes for procedures and other documents, as well as routing and managing YMP deliverables. On a more specific level, technical writers and editors help authors create, write, review, and edit many types of documents; in addition, the technical information center staff identify and locate reliable references to ensure the defensibility of such documents. An authorized derivative classifier reviews the documents prior to publication. Document control issues approved documents for use across this program, while web publishing prepares documents for deployment to the internet. These activities help to implement the interim direction on document development. Services are performed in compliance with the requirements of the *Quality Assurance Requirements and Description* (DOE 1998a) and applicable DOE and NRC regulatory documents.

- Records management services provides for capturing, indexing, imaging, maintaining, storing, retrieving, and finally, archiving DOE records, which are required for the licensing process. The records are currently available through an electronic indexing system pointing to electronic images available for printing. Full-text search capabilities have been added to support development of the LA. Support is also provided for responding to requests filed under the Freedom of Information Act (1997) and performing specialized records searches for DOE personnel. These activities are crucial for capturing the information that must eventually be accessible through a licensing support network, as required by NRC in 10 CFR 2, Subpart J, to support NRC licensing proceedings. Records liaisons assist designated records coordinators in line organizations with training and assistance in submitting correct records packages. DOE records are maintained according to the requirements of DOE Order 0 200.1, Information Management Program, DOE Guide G 1324.5B, Implementation Guide for Use with 36 CFR Chapter XII - Subchapter B Records Management, Records Management Program, and the *Quality Assurance Requirements and Description for the Civilian Radioactive Waste Program* (DOE 1998a), Section 17. Mail and action-item tracking services are also provided.

- DOE and NRC requirements, as codified in the CFR, are implemented for the configuration management of both physical and software assets acquired or developed for DOE. DOE Order 430.1, Life Cycle Asset Management, is followed to maintain an acceptable level of configuration integrity in both design and acquisition for any physical assets acquired or built for DOE. This service also plans and monitors DOE controls, including baseline change control, change control thresholds, and status. Thus, physical integrity is ensured for physical assets and systems during the operations and maintenance phase. Technical guidance on configuration management of software is derived from NUREG-0856, *Final Technical Position on Documentation of Computer Codes for High-Level Waste Management* (NRC 1983).

- Copyrighted reference materials cited within DOE documents or records are acquired, maintained, and indexed into a searchable database in an OCRWM technical information center.

- Technical databases for quality assured licensing data required by DOE are collected, maintained, and stored in databases developed specifically to provide site-specific and regional geoscience-characteristic data needed to analyze the design of the repository system, to generate performance assessment findings, and to assess the environmental impact. Three system components develop and maintain four relational databases. An automated technical data-tracking component creates the reference information database. A geographic nodal information study and evaluation system creates and retrieves data from a site and engineering properties database. Together, a geographic information database and a reference information database compile the latest available data about site and regional characteristics into a database of summarized and interpreted characteristics.

5.2.3 Systems Engineering

The proposed Monitored Geologic Repository is a large, technically complex combination of scientific, engineering, and construction activities. Systems engineering is used to coordinate and integrate these diverse activities to ensure that site characterization activities and the LA design meet the regulatory requirements for safely containing waste from the public. These requirements are found in the Nuclear Waste Policy Act of 1982, as amended, and in requirements written by DOE, NRC, and EPA.

Systems engineering plays four major roles in developing a safe and effective geologic repository:

- The first role is to translate regulatory requirements into design criteria so the repository itself and the waste packages can be designed, specified, and constructed to meet these requirements. Guiding the implementation of those requirements is the need to relate specific site-characterization activities and specific components of the repository, as currently designed, to their potential impact on radiological safety and waste containment. In addition, plans for safeguarding nuclear materials and for protecting against radiological sabotage will be prepared for the LA.

- The second role of systems engineering is to review the repository and waste package designs as they progress from the conceptual through the preliminary and on to the final stages. Such review ensures that the designs included in the LA meet regulatory requirements and are properly integrated throughout the design process. This activity is referred to as requirements and design verification.

- The third role is to assure DOE and NRC that the repository and waste packages, if built as designed, will operate in a safe, efficient, and cost-effective manner and that design and specification changes are documented and controlled in accordance with the approved quality assurance program.

- The fourth role is to develop an operations test program to demonstrate that the designs will operate as intended. This program, intended to test the operations of the actual facility, is developed during the design process and continues through the start-up phase.

Successfully applying the principles of systems engineering ensures that performance of a monitored geologic repository, as detailed in the LA, will be balanced against construction and operation costs of the repository. This is accomplished by cost/benefit analyses, design alternative evaluations, engineering trade-off studies, and operations analyses that support the development of the design and cost estimates. These activities are performed throughout the life of YMP to provide the proper balance between repository performance and cost, and are documented to provide the technical bases that support the NRC licensing process.

5.2.4 Quality Assurance Program

The DOE quality assurance program serves to ensure the quality of data, analyses, and materials that are related to structures, systems, or components important to radiological safety or waste isolation. The quality assurance program satisfies NRC requirements in 10 CFR 60, Subpart G, and has been audited and accepted by NRC. This program requires that all quality-affecting work be performed according to written and approved procedures. It prescribes requirements for documentation of quality-affecting work and employs audits and surveillances to ensure the quality of such work and of safety-related materials.

The quality assurance program applies to data collection in the field, laboratory testing, model development, and design processes. It requires documentation that individuals are educated, experienced, and qualified to perform assigned work; it also requires that they be trained in applicable procedures. The quality assurance program requires documentation of this work so that an independent reviewer can trace how, where, and when the work was performed.

The quality assurance program applies to all organizations that do quality-affecting work, including DOE, contractors, and vendors who supply goods and services. All quality-affecting work is subject to audits and surveillances by the Office of Quality Assurance. The objective of the audits and surveillances is to verify that all quality assurance requirements are being met, to allow early identification and correction of problems, and to provide information that managers can use to improve management processes.

5.2.5 Project Management and Control

Project management, control, and integration activities support development of the LA in the following ways:

- Providing project management for major participants

- Maintaining and operating project-level planning and control systems to support planning and performance measurement in accordance with relevant DOE Orders and applicable industry practices

- Maintaining the work breakdown structure index and dictionary

- Monitoring performance and controlling affected organization activities

- Providing cost and schedule performance reports

- Supporting periodic management reviews, program reviews, midyear reviews, and responses to stakeholder and program oversight inquiries concerning DOE plans and progress

In addition, as part of the LA, project management will be developing an organizational, management, and staffing structure that can support construction and operation of a monitored geologic repository.

5.2.6 Institutional and External Affairs

Stakeholders are informed about YMP through various products, communication services, and activities. Three areas are covered in this program.

The first area, intergovernmental relations, provides the following services:

- Planning and coordinating interactions with local, state, tribal, and federal government entities, regulatory agencies and oversight bodies, the utility industry, environmental organizations, the Nevada business community, other stakeholders, and the public

- Coordinating the County Representative Program, which includes meeting with representatives of the 10 counties directly affected by YMP and with officials of the State of Nevada, meeting with citizens in rural areas, and providing informational materials

- Providing public policy and legislative analyses about Yucca Mountain-related issues at the congressional, state, and local levels

- Creating opportunities for public involvement, when appropriate, in decision-making processes concerning YMP

- Supporting media interactions, responding to media inquiries, preparing press releases, and escorting media representatives on onsite tours

- Conducting communication workshops designed to enhance the staff's public speaking skills and increase effectiveness of media interactions

- Producing videos about different aspects of YMP

- Maintaining a news clipping service with program-wide distribution. This is a computerized video and print database that includes specialty items and news footage

The second area is public outreach, which includes the following:

- Conducting tours of Yucca Mountain for congressional representatives and other government officials, regulators, utility representatives, the media, students, and the general public

- Operating science centers in Las Vegas, Pahrump, and Beatty, Nevada, to provide residents of these communities with information about YMP scientific, engineering, and technical activities and accomplishments

- Providing a speakers' bureau of YMP employees who give presentations to community and educational organizations on a wide variety of topics related to the disposal of spent nuclear fuel and high-level radioactive waste

- Staffing exhibits at technical conferences and community events to provide updates on YMP activities and milestones

- Maintaining the YMP internet web site to provide in-depth information about site characterization efforts and provide a convenient medium for the public to ask questions and provide comments about YMP-related issues and activities

- Operating a national, toll-free information line as part of a program to disseminate information to the public and respond to queries

- Developing educational programs designed to assist the public in making informed decisions about issues related to YMP

- Operating the OCRWM publications distribution center, which sends publications and other materials about YMP to requestors worldwide

The third area focuses on product development, which includes tangible communications products involving the following:

- Articulating the YMP strategic identity by encapsulating the image of the YMP via forms of fixed media (written, audio, visual)

- Developing communications such as fact sheets, brochures, information sheets, documents, exhibits, and videos as a means to provide information to the public about YMP scientific, engineering, and technical activities and accomplishments

- Designing the Yucca Mountain science centers to ensure that information about the YMP is easily accessible, understandable, and versatile enough to appeal to a broad spectrum of interest and knowledge levels

- Providing newspaper advertisements, radio announcements, and informational flyers in support of ongoing YMP public outreach and stakeholder involvement activities

- Disseminating information products to both internal and external parties

5.2.7 Administrative Services

The administrative support staff manages the DOE physical plant, property, and contracts as required to complete the characterization of the Yucca Mountain site.

These activities are performed in accordance with federal, state, and local laws and regulations; DOE orders; and accepted industry practices. These activities include the following:

- Procuring goods and services

- Administering contracts

- Managing property

- Administering, security, personnel, and publications services

- Administering building and vehicle leases

- Providing utilities and telecommunications

- Planning the use of space

- Maintaining and modifying buildings

- Facilitating office moves

- Providing security for YMP Area 25 of the Nevada Test Site

The administrative support staff also provides some of the above services to the USGS and non-M&O entities associated with YMP.

5.2.8 Training Support

The YMP training mission is to ensure that personnel are properly trained before performing their work. Effective training support takes a systematic approach comprised of five fundamental activities:

- Identifying and analyzing training needs
- Designing training based on those needs
- Developing training
- Implementing training
- Evaluating training

DOE and NRC recognize that this systematic approach meets the requirements for training at facilities under their direction and oversight. This approach to training will be implemented in the YMP training program before completion of the

LA, as required by 10 CFR 60, Subpart H, and 10 CFR 60.21(c)(15)(iii). The training program will be described in the LA and, when in place, will fully support construction and operation of the facility.

Following the training mission's directive, the training staff track the training of all personnel, notifying them when required training is offered. The staff processes all personnel training and qualification records into the records information system, and provides scheduling, registration, and logistics support for all training.

The training section of the safety analysis report, which will be submitted with the LA, describes the qualification/certification requirements for personnel in management, operations, and maintenance. Training and qualification activities satisfy the requirements of the DOE quality assurance program. These activities are conducted in accordance with federal, state, and local laws and regulations, as well as standard industry practices.

Project personnel are trained, as required, in the following areas:

- Project orientation
- Environmental protection
- Radiological protection
- Safety and health
- Quality assurance
- Industrial hygiene
- Underground operations
- Emergency management
- Work processes compliance

Project personnel are also trained on conduct of operations. This training ensures achievement of a high level of performance in facility operations through effective use of good work practices. These involve plans, procedures, and directives to implement and control operations activities.

Performance-based training techniques are used to develop training based on the tasks that a person actually performs. This technique, along with the overall systematic approach to training, provides DOE management with a proven mechanism to adjust qualification requirements as task assignments change or new equipment is installed.

6. COSTS

YMP activities are summarized in the program elements listed below. Sections within this volume, noted parenthetically in the list, provide information about work in the program elements that supports development of a license application for repository construction. The cost estimates summarized for these program elements are shown in Table 6-1. Cost estimates for YMP activities comprising these program elements are shown in Table 6-2. Refer to Figure 7-2 for the schedule of those activities. See Table 7-1 for list of acronyms and abbreviations used in Tables 6-1 and 6-2.

- Site investigations work (Section 3.1)
- Design work (Section 3.2)
- Performance assessment work (Section 3.3)
- EIS and environmental compliance (Section 4.1)
- Site recommendation (Section 4.2)
- Licensing (Section 4.3)
- Field construction and operations activities (Section 5.1)
- Other support activities (Section 5.2)

The "Financial & Technical Assistance" category includes two cost components:

- Technical and financial assistance provided to the State of Nevada and affected units of local government
- Payments equal to taxes made to the State of Nevada and any affected unit of local government in an amount each fiscal year equal to the amount the state or affected unit of local government would receive if authorized to tax characterization activities at the site

Because these costs are not associated with scheduled activities, this category has no schedule entry in Figure 7-2.

Table 6-1. Summary of Estimated Costs to Submittal of License Application
($000,000 in year of expenditure)

Program Element	FY1999	FY2000	FY2001	FY2002[1]	Total
Site Investigations	59.9	62.0	52.9	14.4	189.2
Design	77.3	84.5	90.2	44.5	296.5
Performance Assessment	19.5	19.0	18.3	6.9	63.6
EIS & Environmental Compliance	20.8	22.1	14.3	6.9	64.1
Site Recommendation	0.9	0.9	1.1	0	2.9
Licensing	20.1	24.3	22.5	9.7	76.6
Field Construction & Operations Activities	31.0	36.2	27.5	11.4	106.1
Other Support Activities[2]	66.9	87.2	87.5	35.7	277.3
Financial & Technical Assistance	11.6	22.3	20.0	7.9	61.8
TOTAL YMP	**308.0**	**358.5**	**334.2**	**137.4**	**1138.1**

[1]Partial year costs through submittal of LA on March 1, 2002.

[2]This element contains the escalation adjustment.

Table 6-2. YMP Estimated Costs to Submittal of License Application
($000)

ACTIVITY ID	DESCRIPTION	FY 1999	FY 2000	FY 2001	FY 2002
Site Investigations					
2270	ST205 - Single Heater Test Cool Down	220	0	0	0
2025	Data/Analy Addr Seepg-UZ Flow/Trnspt-TSPA-SR/LA	1826	150	0	0
2029	Data/Analy Eval Dilution Pathways-SZ for TSPA-SR/LA	1164	75	0	0
2033	NFE Results to support TSPA-SR	2226	0	0	0
2035	NFE Results to Eval WP Life & EBS Trnspt for SR/LA	2419	1329	0	0
2027	Modeling to Eval Seepg & UZ Flow/Trnspt - TSPA-SR/LA	2096	1003	0	0
2031	Modeling Eval Dilution & Pathways in SZ for TSPA-SR/LA	1543	692	0	0
2210	SR-Develop Hydrog Frmwork & Eval Disruptive Events	2683	1483	0	0
2050	Testing for Enhanced Characteriz of Repos Block	2907	5135	1214	0
6107	ST215 Drift Scale Heater Test - Heat Up Phase	4708	4212	4855	1054
6105	Science Support to License Application	3882	3647	4645	947
SANC150*	Nye County Drilling & Monitoring	4669	4669	4579	1627
7027	Performance Confirmation Data Collection	388	1692	2339	755
8621	Test Coordination/Support for Site Activities	5633	4212	2792	727
9090	IN03X - Site Investigations Base Support	12794	12125	10349	1770
2215	Data/Analy Update Seepg & UZ Flow/Trnspt-SR/LA	3201	2475	0	0
2245	Data/Analy Update Dilution Pthwys in SZ for SR/LA	1004	3722	0	0
2253	NFE Data and Analysis Update Support SR/LA	923	823	0	0
8615	Conf NFE Modeling Concl for Constr Authorization	0	2264	3398	835
8607	Modlng Conf Concl Seepg & UZ Flow/Trnspt- CA	0	842	1214	257
8611	Data/Analy Conf Concl Dilut Pthwys - SZ TSPA-SR/LA	0	842	1214	343
8601	Confirm Hydrog Frmwrk/Eval Disrupt Events for CA	0	0	243	0
8605	Data/Analy Conf Rslts - Seepg & UZ Flow/Trnspt-CA	0	0	2063	406
8609	Data/Analy Conf Concl Dilution Pthwys-SZ for CA	0	0	2792	600
8613	Conf NFE Data Rslts for Constr Authorization (CA)	0	0	485	128
SASA108*	AECL Consultation	600	600	689	244
SASA195*	Nevada University Scientific Studies	5000	10000	10000	4063
7035	Drift Scale Heater Test - Cool Down Phase	0	0	0	601
	TOTAL	$59,886	$61,992	$52,870	$14,357
Design					
2382	Subsurface Initial Design & Alternatives/Options	680	392	0	0
2375	WP Initial Designs and Options for SR/LA	1574	0	0	0
2021	Alternatives/Options Evaluations	547	0	0	0
2310	LA Requirements Development	6782	5189	0	0
7040	Long Term WP Mtls Testing and Modeling for SR/LA	11230	10876	3902	359
7030	WP122 - Long-Term Waste Form Testing & Modeling	6291	5912	882	0
2023	LA Design Integration/Verification	456	970	973	0
2403	SR Design	3011	11936	1184	0
2380	Neutronics Methodology Development	1364	948	970	843
2371	WP Input for Licensing, PA, SR, PC & EIS	1300	909	606	526
2408	LA Submittal Support	475	285	329	107
2373	WP Closure Weld Designs and Examination Developm	956	2115	1216	429
2383	Complete Proposed SR/LA Design	6289	22219	2000	0
2475	Interface Configuration Management	1603	1685	1941	683
7003	PC - Test Development & Definition	2042	2006	978	968
9075	INO1X - Systems Engineering Base Support	6354	6423	7396	2608
9085	IN02X - Waste Package Base Support	3020	3817	3672	1427

Table 6-2. YMP Estimated Costs to Submittal of License Application ($000) (Continued)

ACTIVITY ID	DESCRIPTION	FY 1999	FY 2000	FY 2001	FY 2002
9195	Total System Life Cycle Cost Estimation	891	711	728	634
2377	WP Proposed Designs for SR/LA	2580	5034	0	0
2315	Develop Procure & Construct Requirements	0	939	3881	5153
2024	Final SR/LA Design and Verification	0	100	196	0
8327	Procurement & Construction Design	0	0	10843	8261
6109	Subsurface Prototype System Design	0	0	2300	847
6113	WP Design Completion	0	639	5160	3410
7032	WP123 - Long Term Waste Form Testing and Modeling	0	0	5127	2359
7042	Long Term WP Materials Testing and Modeling	0	0	9097	4001
8407	Preliminary Design (Balance of Plant)	0	0	5702	1806
8360	Procure & Construct Subsurface Design	0	171	20852	9710
2390	WP Reference Design	1426	500	0	0
2391	WP Alternatives/Evaluations for SR/LA	1111	0	0	0
2395	Subsurface Alternatives/Enhancements for SR/LA	190	0	0	0
2387	Subsurface Reference Design	11010	245	0	0
2392	Surface Reference Design	3213	519	0	0
2393	Surface Alternative/Enhancements Design	1712	0	0	0
2394	Surface Initial Designs/Options for SR/LA	1216	0	0	0
7063	Procure/Constr Design Integration/Verification	0	0	243	327
	TOTAL	$77,323	$84,540	$90,178	$44,459
Performance Assessment					
1122	TSPA-VA Documentation	1364	0	0	0
2175	Develop Abstract/Test Disruptive Events	1159	627	0	0
2176	Develop Abstract/Test SZ and Biosphere	1469	1148	0	0
2190	Develop Abstract/Test Waste Form & EBS Transport	2104	942	0	0
2195	Develop Abstract/Test of WP Degradation	895	366	0	0
2220	Develop Abstract/Test of UZ Flow & Tranport	2834	1819	0	0
2235	Develop Abstract/Test of Near Field Environment	3049	1586	0	0
2184	Process Control & Management	735	737	840	215
2185	Design Analyses	1991	850	485	128
7005	Regulatory Analyses	1259	737	727	3422
2396	TSPA Approach & Model Development	2309	3730	0	0
2397	TSPA for SR/LA	0	5650	1472	0
2398	TSPA Support/Sensitivity Studies	0	0	9195	0
2399	Finalize TSPA SR/LA Analyses	0	0	5254	3052
8201	EIS Analyses	315	842	364	128
	TOTAL	$19,483	$19,034	$18,345	$6,946
EIS & Environmental Compliance					
3040	NE1024 - DEIS Prepare and Issue	2204	0	0	0
9121	IN08X1 - Environment, Safety & Health Core Program	12200	13881	13056	6853
3070	NE1041 - FEIS Prepare and Issue	420	2633	1057	0
SASA176*	EIS Contractor	5938	5583	225	0
	TOTAL	$20,762	$22,097	$14,338	$6,853
Site Recommendation					
2019	Site Recommendation Support	892	895	1057	0
	TOTAL	$892	$895	$1,057	$0

Table 6-2. YMP Estimated Costs to Submittal of License Application ($000) (Continued)

ACTIVITY ID	DESCRIPTION	FY 1999	FY 2000	FY 2001	FY 2002
Licensing					
2115	LA200 - Prepare Working Draft LA	2113	110	0	0
6101	Licensing Case Development	945	632	0	0
2570	Site Characterization Progress Reports	210	211	211	0
6260	ES & H Licensing Support	2587	2880	2009	385
2085	Table of Contents and Indices Preparation	157	406	408	165
2186	Regulatory Support	859	830	423	230
2187	Regulatory Interactions	1046	1166	1243	872
2188	Regulatory Reviews	744	751	793	400
2189	Commitment Management	503	560	608	329
2191	Regulatory Counsel and Consultation	1391	1580	1850	1155
2470	Technical Data Management	8620	10084	8751	3579
2600	Regulatory and Licensing Integration	915	1673	1877	1006
6103	Complete License Application	0	3377	4319	1555
	TOTAL	$20,090	$24,260	$22,491	$9,676
Field Construction & Operations Activities					
2017	ES3010 - Excavate East-West Drift	0	2099	0	0
2450	ES088 - Prepare for Long Term ESF Operations	4666	0	0	0
9103	IN06X - ESF Construction Support	16293	23951	17235	6960
9105	IN07X - Site Facilities Management	10076	10109	9725	3933
8451	RFP Final Preparation for Subsurface Contractor	0	0	502	246
8453	RFP Final Preparation for Surface Contractors	0	0	54	288
	TOTAL	$31,035	$36,159	$27,516	$11,428
Other Support Activities					
2415	IS140 - Records Processing for SR/LA	574	2580	0	0
SASA205*	NAS Consultation	37	37	37	0
SASA206*	University of Nevada Reno Programs	140	140	100	0
USBOR*	U.S. Bureau of Reclamation Services	587	587	587	0
2165	LA900 - Web Based Information System	1793	2036	1979	997
2480	Information Management Support of LA Information	2246	3111	1163	469
9110	IN15X - Administrative Support	3207	3268	3268	1327
9111	IN15X - Training Support	2847	3057	3056	1241
9125	IN14X - Institutional Interactions	3798	3791	3806	1542
9130	Information Technology	8639	11599	10043	4064
9135	IN09X - Project Management	7883	7908	7868	3156
9197	IN12X - Program Information Management	9016	8214	8034	3250
CLOSE*	Subcontract Close Out	1905	2003	1995	816
SASA115*	Administrative Support Contractor	871	836	965	393
SASA125*	Information Management Services Contractor	2428	2330	2690	1097
SASA145*	Security Services Contractor	668	668	668	273
SASA185*	Quality Assurance Services Contractor	8603	8603	8577	3500
SASA225*	Telecommunications Services	1875	1875	1875	761
SASA235*	Televideo Services	150	150	150	60
ESCALATE*	Escalation	0	7462	15424	8468
SASCORE2*	Lease Scoring	0	7700	4600	0
9913*	Management & Technical Services Contractor	8973	8609	9970	4068
BMSA308*	US Geological Survey Management	630	630	630	254
	TOTAL	$66,870	$87,194	$87,484	$35,737

Table 6-2. YMP Estimated Costs to Submittal of License Application ($000) (Continued)

ACTIVITY ID	DESCRIPTION	FY 1999	FY 2000	FY 2001	FY 2002
Financial & Technical Assistance					
SASA155*	Financial & Technical Assistance	5930	12329	9950	3871
SASA165*	Payments Equal to Taxes	5729	10000	10000	4063
	TOTAL	$11,659	$22,329	$19,950	$7,934
	REPORT TOTAL	**$308,000**	**$358,500**	**$334,231**	**$137,390**

*These support activities and cost elements are not shown on the schedules in Section 7.

Note: Reported total dollars are for the year of expenditure.

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7. SCHEDULE

The activities of the YMP are scheduled in detail to integrate and manage the following project objectives:

- Publish a draft EIS in fiscal year 1999.
- Complete the final EIS in fiscal year 2000.
- If the site is suitable for development as a repository, complete the Secretary of Energy's site recommendation to the President, accompanied by the final EIS, in fiscal year 2001.
- If the President recommends the site to Congress, and the designation becomes effective, submit an LA to NRC in fiscal year 2002.

If these objectives are achieved, formal licensing interactions with NRC will occur through 2005 or beyond, leading to a repository construction authorization. These interactions will include formal regulatory hearings, and may require provision of additional data and analyses. Work will also continue on developing the final design for construction, collecting data for performance confirmation, and updating the performance assessment.

Key YMP milestones are summarized in Figure 7-1 and are excerpted from the YMP proposed project summary schedule shown in Figure 7-2. Refer to Section 6 for the costs associated with this schedule. Additional acronyms and abbreviations used in the schedules are defined in Table 7-1.

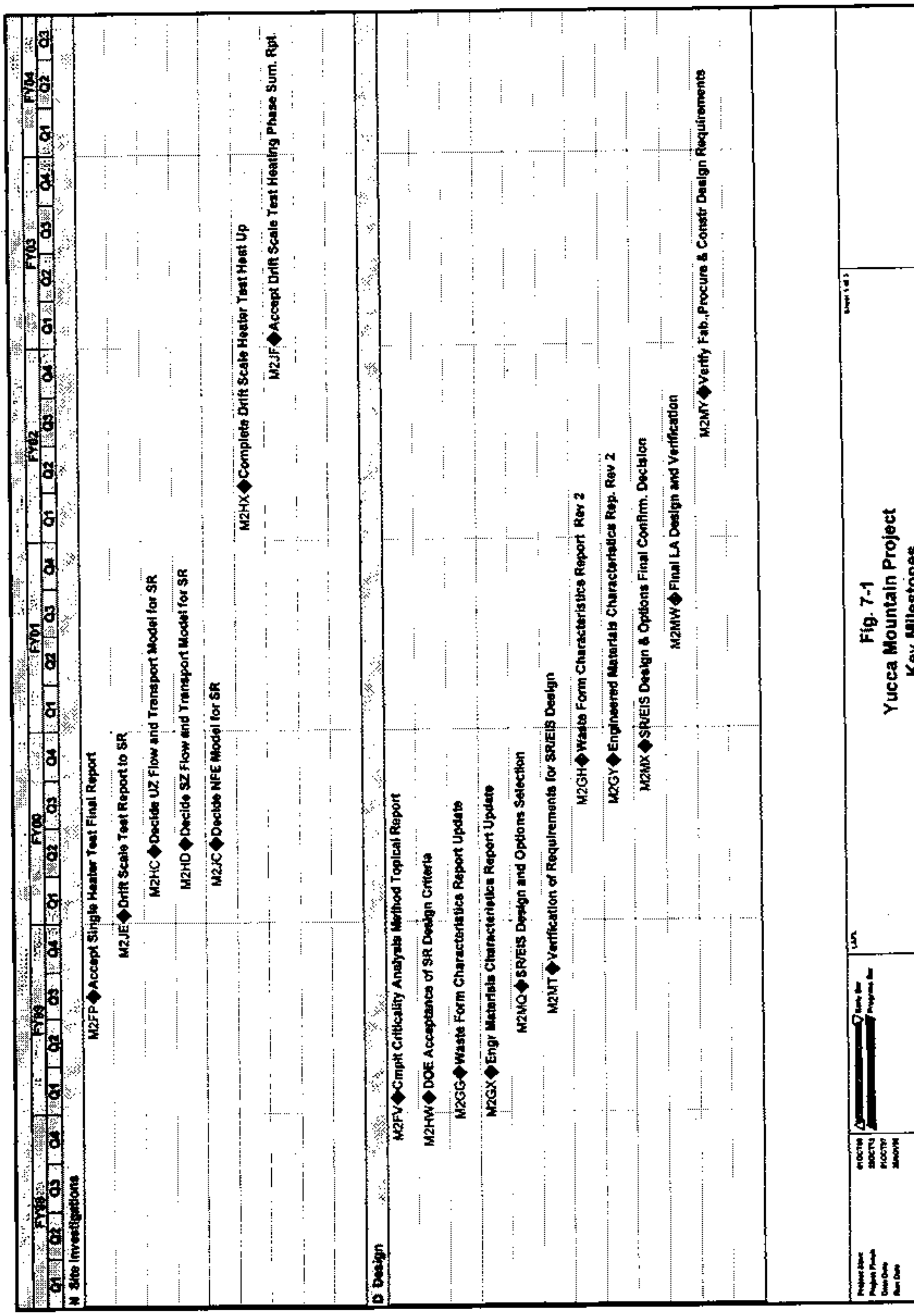


Figure 7-1. YMP Key Milestones

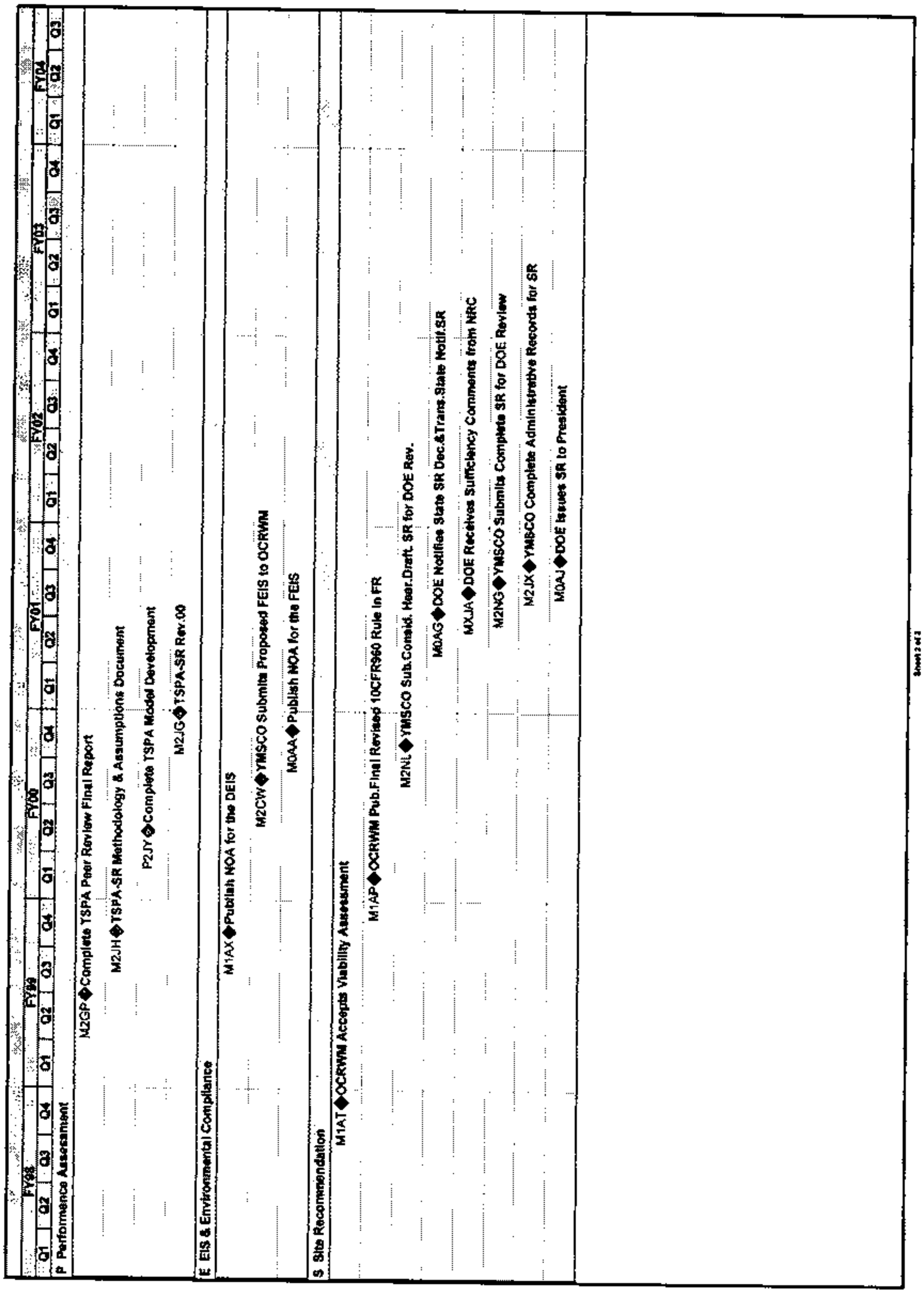


Figure 7-1. YMP Key Milestones (Continued)

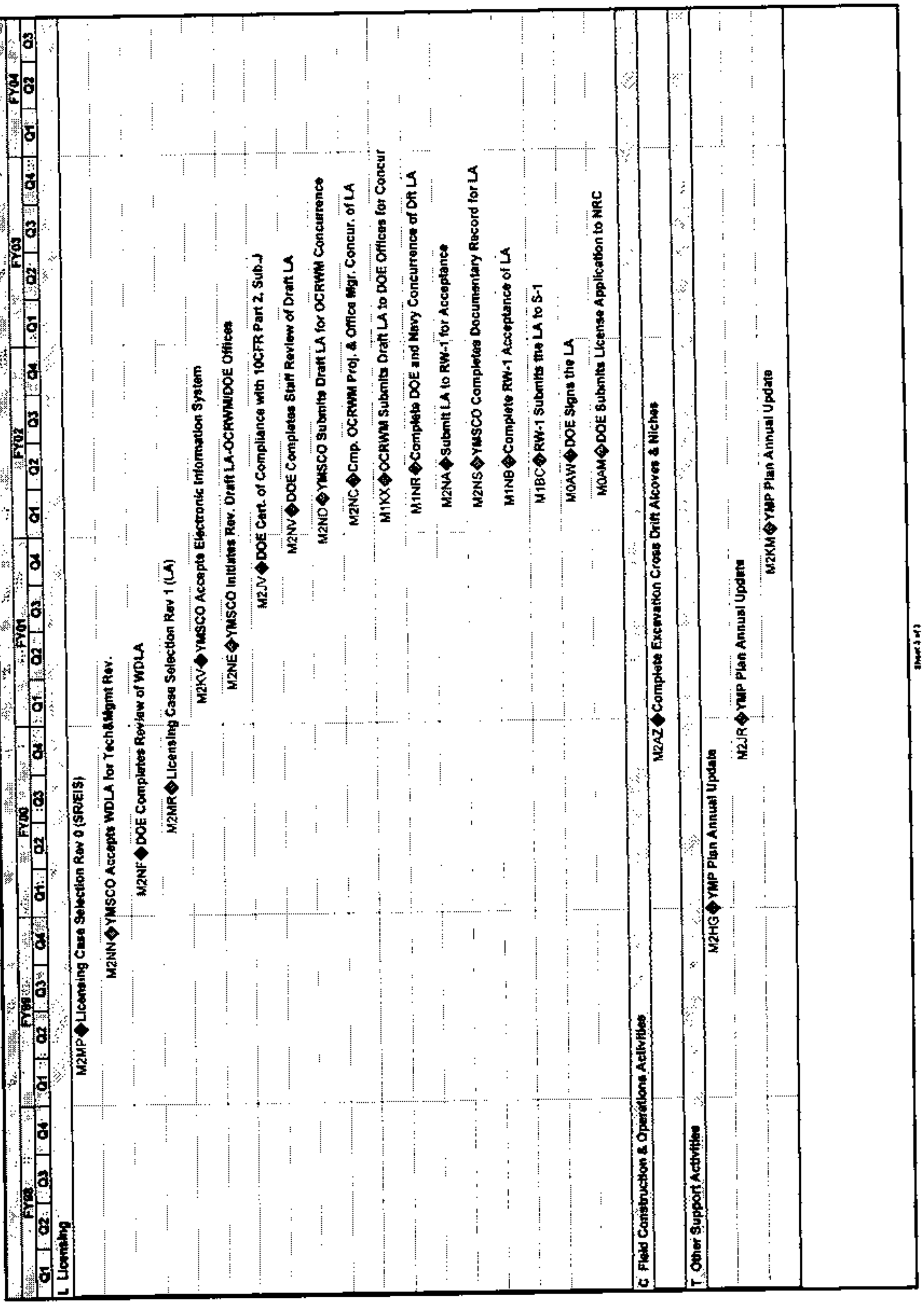


Figure 7-1. YMP Key Milestones (Continued)

Activity ID	Activity Description	Early Start	Early Finish
N Site Investigations			
2270	ST205 - Single Heater Test Cool Down	01OCT98*	15JAN99
2033	NFE Results to support TSPA-SR	01OCT98*	16JUN99
2025	Data/Analy Addr Seepg-UZ Flow/Trnspt-TSPA-SR	01OCT98*	28DEC99
2029	Data/Analy Eval Dilution Pathways-SZ for TSPA-SR	01OCT98*	28DEC99
2035	NFE Results to Eval WP Life/EBS Trnspt for SR/LA	01OCT98*	14JAN00
2027	Modeling to Eval Seepg/UZ Flow/Trnspt - TSPA-SR	01OCT98*	01FEB00
2031	Modeling Eval Dilution/Pathwys in SZ for TSPA-SR	01OCT98*	15FEB00
2210	SR-Develop Hydrog Frmwork/Eval Disruptive Events	01OCT98*	29SEP00
2050	Testing for Enhanced Characteriz of Repos Block	01OCT98*	28SEP01
6107	ST215 Drift Scale Heater Test - Heat up Phase	01OCT98*	09JAN02
6105	Science Support to License Application	01OCT98*	28FEB02
7027	Performance Confirmation Data Collection	01OCT98*	31MAR05
8621	Test Coordination/Support for Site Activities	01OCT98*	31MAR05
9090	IN03X - Site Investigations Base Support	01OCT98*	31MAR05
2215	Data/Analy Update- Seepage/UZ Flow/Trnspt-SR/LA	16FEB99*	29SEP00
2245	Data/Analy Update DilutionPthwys in SZ for SR/LA	16APR99*	29SEP00
M2FP	Accept Single Heater Test Final Report		14MAY99*
2253	NFE Data and Analysis Update Support SR/LA	16JUN99*	29SEP00
M2JE	Drift Scale Test Report to SR		15OCT99*
8616	Conf NFE Modeling Concl for Constr Authorization	17JAN00*	28FEB03
8607	Modeling Conf Concl Seepg & UZ Flow/Trnspt- CA	01FEB00*	31DEC02
M2HC	Decide UZ Flow and Transport Model for SR		15FEB00*
8611	Data/Analy Conf Concl Dilut Pthwys - SZ TSPA-SR	16FEB00*	31DEC02
M2HD	Decide SZ Flow and Transport Model for SR		29FEB00*
M2JC	Decide NFE Model for SR		29FEB00*
8601	Confirm Hydrog Frmwrk/Eval Disrupt Events for CA	02OCT00*	28SEP01
8606	Data/Analy Conf Rslts-Seepg & UZ Flow/Trnspt-CA	02OCT00*	31DEC02
8609	Data/Analy Conf Concl Dilution Pthwys-SZ for CA	02OCT00*	31DEC02
8613	Conf NFEData Rslts for Constr Authorization (CA)	02OCT00*	28FEB03
M2HX	Complete Drift Scale Heater Test Heat Up		09JAN02*
7035	Drift Scale Heater Test - Cool Down Phase	10JAN02*	10JAN06

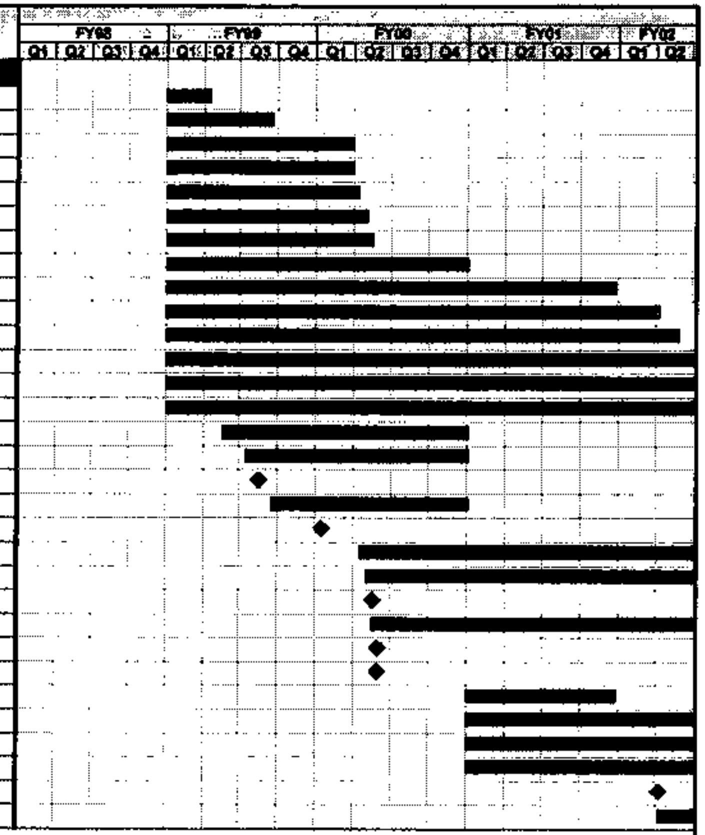


Fig. 7-2
Yucca Mountain Project
Proposed Project Summary Schedule

Figure 7-2. YMP Proposed Project Summary Schedule

Activity ID	Activity Description	Early Start	Early Finish
D Design			
2382	Sub-Surface Initial Design/Alternatives/Options	01JUL98	30JUN00
2375	WP Initial Designs and Options for SR/LA	01OCT98*	27MAY99
2391	WP Alternatives Evaluations for SR/LA	01OCT98*	27MAY99
2395	Sub-surface Alternatives/Enhancements for SR/LA	01OCT98	27MAY99
2021	Alternatives/Options Evaluations	01OCT98*	28MAY99
2393	Surface Alternative/Enhancements Design	01OCT98*	28MAY99
2394	Surface Initial Designs/Options for SR/LA	01OCT98*	28MAY99
2387	Sub-Surface Reference Design	01OCT98*	30DEC99
2392	Surface Reference Design	01OCT98*	30DEC99
2310	LA Requirements Development	01OCT98	01MAR00
2390	WP Reference Design	01OCT98*	30JUN00
7030	WP122 - Long-Term Waste Form Testing & Modeling	01OCT98*	31OCT00
2023	LA Design Integration/Verification	01OCT98	01NOV00
7040	Long Term WP Mtls Testing and Modeling for SR/LA	01OCT98*	31OCT01
2380	Neutronics Methodology Development	01OCT98*	28FEB02
2371	WP Input for Licensing, PA, SR, PC & EIS	01OCT98*	01MAR02
2408	LA Submittal Support	01OCT98*	01MAR02
2373	WP Closure Weld Designs and Examination Developm	01OCT98*	30SEP02
2475	Interface Configuration Management	01OCT98*	30SEP02
7003	PC - Test Development & Definition	01OCT98*	30SEP02
9075	6101	01OCT98*	30SEP02
9085	IN02X - Waste Package Base Support	01OCT98	30SEP02
9195	Total System Life Cycle Cost Estimation	01OCT98*	30SEP02
M2FV	Cmplt Criticality Analysis Method Topical Report		30OCT98*
M2HW	DOE Acceptance of SR Design Criteria		30OCT98*
M2GG	Waste Form Characteristics Report Update		15DEC98*
M2GX	Engr Materials Characteristics Report Update		15DEC98*
M2MQ	SR/EIS Design and Options Selection		26MAY99*
2377	WP Proposed Designs for SR/LA	28MAY99*	28SEP00
2383	Complete Proposed SR/LA Design	01JUN99	31OCT00
2403	SR Design	01JUN99	01NOV00
M2MT	Verification of Requirements for SR/EIS Design		30JUN99*
2315	Develop Procure & Construct Requirements	01OCT99*	28FEB02
2024	Final SR/LA Design and Verification	01MAR00*	01JUN01
6113	WP Design Completion	03APR00	28FEB02
M2GH	Waste Form Characteristics Report Rev 2		01AUG00*
M2GY	Engineered Materials Characteristics Rep. Rev 2		01AUG00*
M2MX	SR/EIS Design & Options Final Confirm. Decision		25AUG00*
8360	Procure & Construct Subsurface Design	01SEP00	02NOV04
6109	Sub-Surface Prototype System Design	01NOV00*	28FEB02
7032	WP123-Long Term Waste Form Testing and Modeling	01NOV00*	28FEB02

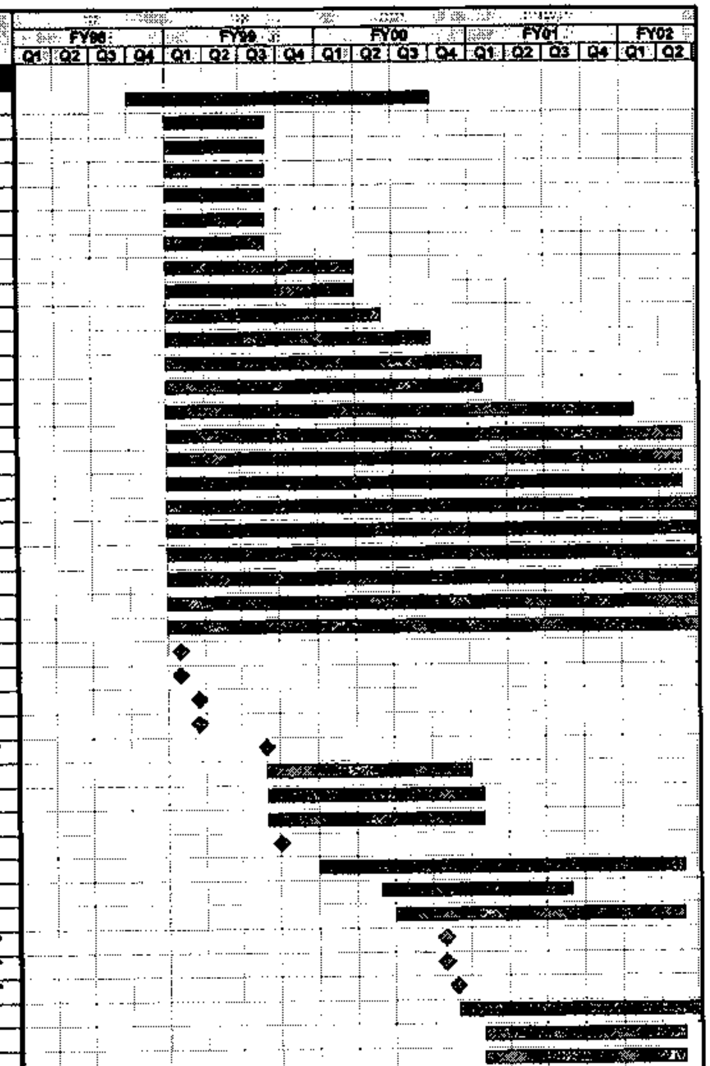


Figure 7-2. YMP Proposed Project Summary Schedule (Continued)

Activity ID	Activity Description	Early Start	Early Finish
7042	Long Term WP Materials Testing and Modeling	01NOV00*	28FEB02
8407	Preliminary Design (Balance Of Plant)	01NOV00*	28FEB03
6327	Procurement & Construction Design	06NOV00*	03NOV05
M2MW	Final LA Design and Verification		01JUN01*
7063	Procure/Constr Design Integration/Verification	02JUL01*	28FEB06
P Performance Assessment			
1122	TSPA-VA Documentation	01OCT98*	30APR99
2396	TSPA Approach & Model Development	01OCT98	29FEB00
2176	Develop Abstract/Test Disruptive Events	01OCT98	31JUL00
2175	Develop Abstract/Test SZ and Biosphere	01OCT98	31JUL00
2190	Develop Abstract/Test Waste Form & EBS Transport	01OCT98	31JUL00
2195	Develop Abstract/Test of WP Degradation	01OCT98	31JUL00
2220	Develop Abstract/Test of UZ Flow & Transport	01OCT98	31JUL00
2235	Develop Abstract/Test of Near Field Environment	01OCT98	31JUL00
2184	Process Control and Management	01OCT98*	30SEP02
2185	Design Analyses	01OCT98*	30SEP02
7005	Regulatory Analyses	01OCT98*	30SEP02
6201	EIS Analyses	01OCT98	30SEP02
M2GP	Complete TSPA Peer Review Final Report		30MAR99*
M2JH	TSPA-SR Methodology & Assumptions Document		30JUL99*
P2JY	Complete TSPA Model Development		11FEB00*
2397	TSPA for SR/LA	01MAR00	31OCT00
M2JG	TSPA-SR Rev.00		29SEP00*
2398	TSPA Support/Sensitivity Studies	01NOV00	31MAY01
2399	Finalize TSPA SR/LA Analyses	01JUN01	30NOV01
E EIS & Environmental Compliance			
3040	NE1024 - DEIS Prepare and Issue	01OCT98*	30JUL99
9121	IN08X1- Environment,Safety & Health Core Program	01OCT98*	28FEB02
M1AX	Publish NOA for the DEIS		29JUL99*
3070	NE1041 - FEIS Prepare and Issue	02AUG99*	28SEP01
M2CW	YMSCO Submits Proposed FEIS to OCRWM		22MAY00*
M0AA	Publish NOA for the FEIS		25AUG00*

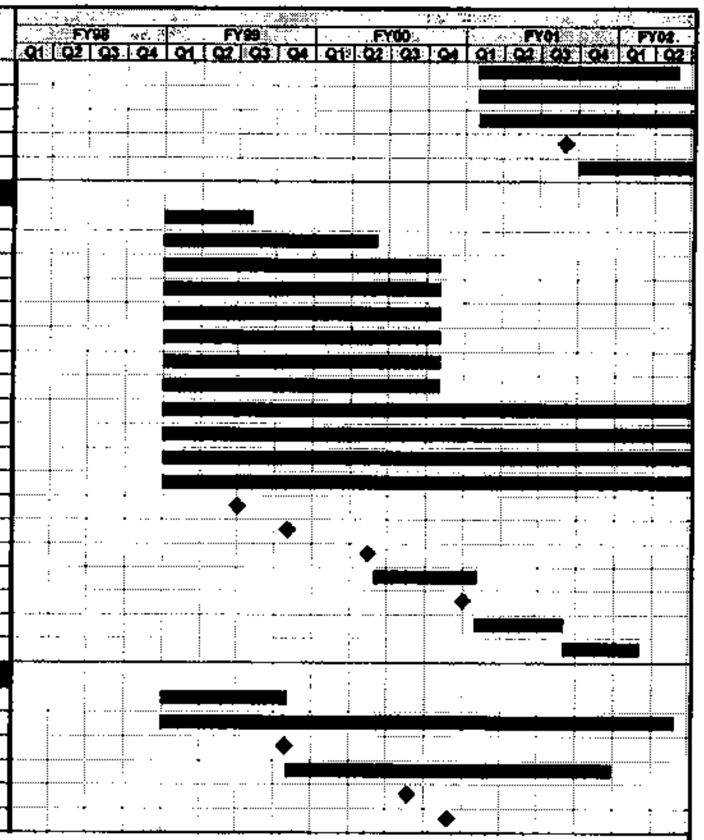


Figure 7-2. YMP Proposed Project Summary Schedule (Continued)

Activity ID	Activity Description	Early Start	Early Finish
S Site Recommendation			
M1AT	OCRWM Accepts Viability Assessment		04SEP98*
2019	Site Recommendation Support	01OCT98*	28AUG01
M1AP	OCRWM Pub.Final Revised 10CFR960 Rule in FR		15NOV99*
M2NL	YMSCO Sub.Consld. Hear.Draft. SR for DOE Rev.		31JUL00*
M0AG	DOE Notifies State SR Dec.&Trans.State Notif.SR		13APR01*
MXJA	DOE Receives Sufficiency Comments from NRC		25MAY01*
M2NG	YMSCO Submits Complete SR for DOE Review		16JUN01*
M2JX	YMSCO Complete Administrative Records for SR		18JUL01*
M0AJ	DOE Issues SR to President		27JUL01*
L Licensing			
2115	LA200 - Prepare Working Draft LA	01OCT98*	28DEC99
6101	Licensing Case Development	01OCT98*	30MAY00
2570	Site Characterization Progress Reports	01OCT98*	30MAR01
6260	ES&H Licensing Support	01OCT98*	28FEB02
2085	Table Of Contents and Indices Preparation	01OCT98*	30SEP02
2186	Regulatory Support	01OCT98*	30SEP02
2187	Regulatory Interactions	01OCT98*	30SEP02
2188	Regulatory Reviews	01OCT98*	30SEP02
2189	Commitment Mangement	01OCT98*	30SEP02
2191	Regulatory Counsel and Consultation	01OCT98*	30SEP02
2470	Technical Data Management	01OCT98*	30SEP02
2600	Strategic Planning, EIS & TSLCC Integration	01OCT98*	30SEP02
M2MP	Licensing Case Selection Rev 0 (SR/EIS)		24FEB99*
M2NN	YMSCO Accepts WDLA for Tech&Mgmt Rev.		27AUG99*
6103	Complete License Application	01OCT99*	30SEP02
M2NF	DOE Completes Review of WDLA		28JAN00*
M2MR	Licensing Case Selection Rev 1 (LA)		01JUN00*
M2KV	YMSCO Accepts Electronic Information System		31JAN01*
M2NE	YMSCO Initiates Rev. Draft LA-OCRWM/DOE Offices		01MAR01*
M2JV	DOE Cert. of Compliance with 10CFR Part 2, Sub.J		27JUL01*
M2NV	DOE Completes Staff Review of Draft LA		15NOV01*
M2ND	YMSCO Submits Draft LA for OCRWM Concurrence		06DEC01*
M2NC	Cmp. OCRWM Proj. & Office Mgr. Concur. of LA		10JAN02*
M1KX	OCRWM Submits Draft LA to DOE Offices for Concur		11JAN02*
M1NR	Complete DOE and Navy Concurrence of Dft LA		31JAN02*
M2NA	Submit LA to RW-1 for Acceptance		07FEB02*
M2NS	YMSCO Completes Documentary Record for LA		08FEB02*
M1NB	Complete RW-1 Acceptance of LA		11FEB02*
M1BC	RW-1 Submits the LA to S-1		13FEB02*
M0AW	DOE Signs the LA		21FEB02*
M0AM	DOE Submits License Application to NRC		01MAR02*

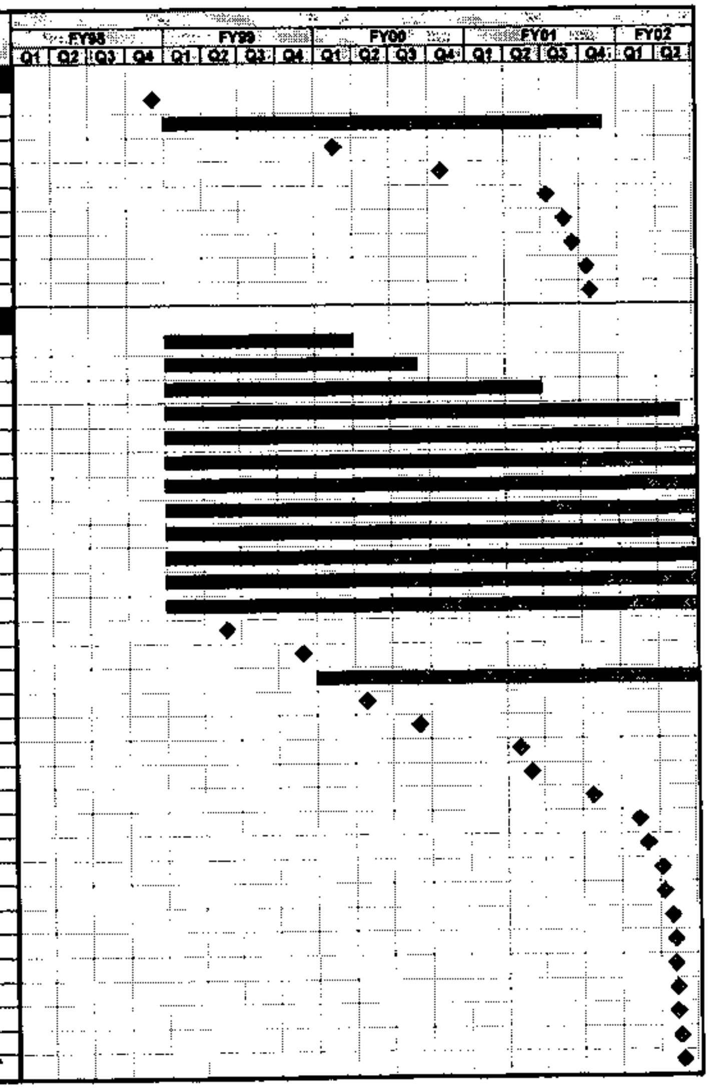
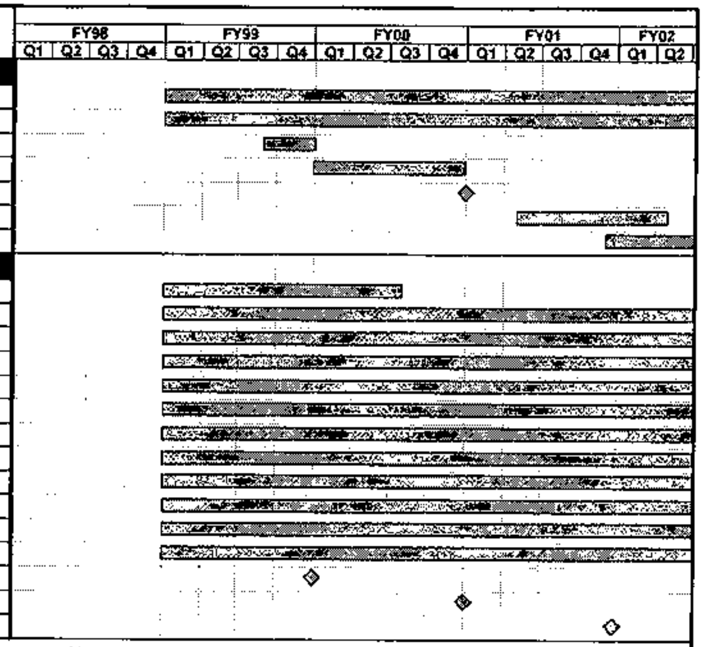


Figure 7-2. YMP Proposed Project Summary Schedule (Continued)

Activity ID	Activity Description	Early Start	Early Finish
C Field Construction & Operations Activities			
9103	IN06X - ESF Construction Support	01OCT98*	30SEP02
9105	IN07X - Site Facilities Management	01OCT98*	30SEP02
2450	ES088 - Prepare for Long Term ESF Operations	01JUN99*	30SEP99
2017	ES3010 - Excavate East-West Drift	28SEP99*	27SEP00
M2AZ	Complete Excavation Cross Drift Alcoves & Niches		29SEP00*
8451	RFP Final Preparation for Subsurface Contractor	01FEB01*	31JAN02
8453	RFP Final Preparation for Surface Contractors	04SEP01*	30AUG02
T Other Support Activities			
2415	IS140 - Records Processing for SR/LA	01OCT98*	28APR00
2165	LA900 - Web Based Information System	01OCT98*	30SEP02
2480	Information Management Support of LA Information	01OCT98*	30SEP02
9110	IN15X - Administrative Support	01OCT98*	30SEP02
9111	IN15X - Training Support	01OCT98*	30SEP02
9120	Set Asides - QA	01OCT98*	30SEP02
9125	IN14X - Institutional Interactions	01OCT98*	30SEP02
9130	Information Technology	01OCT98*	30SEP02
9135	IN09X - Project Management	01OCT98*	30SEP02
9197	IN12X - Program Information Management	01OCT98*	30SEP02
9917	Televideo Support	01OCT98*	30SEP02
9999	Set Asides, MR, Fee and Lease Scoring	01OCT98*	30SEP02
M2HG	YMP Plan Annual Update		30SEP99*
M2JR	YMP Plan Annual Update		29SEP00*
M2KM	YMP Plan Annual Update		28SEP01*



Note:
Milestones identified by four characters beginning with the letter M.
Activities identified with four digit numbers.

Sheet 8 of 8

Figure 7-2. YMP Proposed Project Summary Schedule (Continued)

Table 7-1. Acronyms and Abbreviations for Schedule Milestones and Activities

Abstract	Abstraction
Accept.	Acceptance
Addr	Addressing
AECL	Atomic Energy of Canada Limited
Analy	Analysis
Approv	Approved
CA	Construction Authorization
Cam.	Camera
Cert.	Certification
Characteriz	Characterization
Cmplt	Complete
Concl	Conclusion
Concurren	Concurrence
Conf	Confirm
Constr	Construction
DEIS	Draft Environmental Impact Statement
Dilut	Dilution
Disp	Disposal
Doc.	Document
EBS	Engineered Barrier System
EIS	Environmental Impact Statement
Elect.	Electronic
Engr	Engineering
ESF	Exploratory Studies Facility
ES&H	Environmental Safety and Health
Eval	Evaluation
Fab.	Fabrication
FEIS	Final Environmental Impact Statement
Frmwork	Framework
HQ	Headquarters
Hydrog	Hydrologic
LA	License Application
Mgmt	Management
Modlng	Modeling
Mtls	Materials
NAS	National Academy of Sciences
NFE	Near Field Environment
NRC	Nuclear Regulatory Commission
PA	Performance Assessment
PC	Performance Confirmation
Pthwys	Pathways
Rdy	Ready
Repos	Repository
Rev	Revision
Rpt.	Report
Rslts	Results
RW-1	Organizational symbol for Director, Office of Civilian Radioactive Waste Management

Table 7-1. Acronyms and Abbreviations for Schedule Milestones and Activities (Continued)

Seepg	Seepage
SR	Site Recommendation
Sum.	Summary
Sys.	System
SZ	Saturated Zone
Test	Testing
Top	Topical
Trnspt	Transport
TSLCC	Total System Life Cycle Cost
TSPA	Total System Performance Assessment
U.S.	United States
UZ	Unsaturated Zone
VA	Viability Assessment
WDLA	Working Draft License Application
WP	Waste Package
YMP	Yucca Mountain Project
X1	Symbol for distribution list of other government agencies

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8. REFERENCES

The numbers at the end of each reference are Office of Civilian Radioactive Waste Management document accession numbers. See the inside front cover of this document for whom to contact regarding more information.

8.1 DOCUMENTS CITED

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CRWMS M&O 1997a. *Classification of Permanent Items.* QAP-2-3, Rev. 08. Las Vegas, Nevada: CRWMS M&O. MOL.19980511.0132.

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NRC 1997e. *Issue Resolution Status Report Key Technical Issue: Repository Design and Thermal-Mechanical Effects.* Rev. 00. Washington D.C.: NRC. MOL.19980219.0570.

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8.2 STANDARDS AND REGULATIONS

Unless otherwise dated, the *Codes of Federal Regulations* cited in this document were revised as of January 1, 1998.

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10 CFR 20. Energy: Standards for Protection Against Radiation. 238505.

10 CFR 50. Energy: Domestic Licensing of Production and Utilization Facilities. 238441.

10 CFR 60. Energy: Disposal of High-Level Radioactive Waste in Geologic Repositories. 238445.

10 CFR 72. Energy: Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste. 238589.

36 CFR Chapter XII, Subchapter B. Parks, Forests, and Public Property: Records Management. 238590.

43 CFR 2300. Public Lands: Interior: Land Withdrawals. 238528.

DOE G 1324.5B. Implementation Guide for Use with 36 CFR Chapter XII, Subchapter B: Records Management. 226506.

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DOE Order 430.1. Life Cycle Asset Management. 237625.

DOE Order 470.1 Safeguards and Security Program. 238159.

DOE Order 471.2A. Information Security Program. 238158.

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APPENDIX A

GLOSSARY

APPENDIX A

GLOSSARY

Many of the definitions in this glossary are Yucca Mountain Project specific.

Absorption	The assimilation of dissolved matter in a liquid or gas through the pores or interstices of a solid.
Accessible Environment	(1) The atmosphere, the land surface, surface water, and oceans that humans or animals may contact. (2) The area surrounding a nuclear waste disposal site.
Adsorption	Transfer of solute mass, such as radionuclides, in groundwater to the solid geologic surfaces with which it comes in contact. The term sorption is sometimes used interchangeably with this term.
Advanced Conceptual Design	The design phase that was used to explore selected design alternatives and firmly fix and refine the design criteria and concepts to be made final in later design efforts. The project feasibility was demonstrated, life-cycle costs estimated, preliminary drawings prepared, and a construction schedule developed as required by DOE Order 6410.1.
Advection	A local change in a property, such as temperature or salinity, resulting from flow of a liquid or gas.
Advective Flow	Movement of a fluid because of a pressure gradient.
Advective Transport	Movement of contaminants in a fluid caused by a pressure gradient on that fluid.
Alcoves	Small excavations (rooms) off the main tunnel of a repository that are used for scientific study or for installing equipment.
Alluvium	Sedimentary material deposited by flowing water or by wind.
Aquifer	A subsurface saturated rock unit (formation, group of formations, or part of a formation) of sufficient permeability to transmit groundwater and yield usable quantities of water to wells and springs.
Backfill	(1) The general fill that is placed in the excavated areas of the underground facility. Backfill for the repository may be tuff. (2) The material or process used to refill an excavation.
Barrier	Any material, structure, or condition (as a thermal barrier) that prevents or substantially delays the movement of water or radionuclides.
Baseline Changes	An engineering term meaning a revision of a plan from an established standard or benchmark. A typical change addresses technical scope, cost, and schedule parameters.
Batch Plant	A facility where the components of concrete are mixed.

Biosphere	The ecosystem of the earth and the living organisms inhabiting it.
Borehole	A hole made with a drill, auger, or other tool for exploring strata in search of minerals, supplying water for blasting, emplacing waste, proving the position of old workings or faults, or releasing accumulations of gas or water. Boreholes include core holes, dry-well-monitoring holes, waste-emplacement boreholes, and test holes for geophysical or groundwater characterization.
Bulk Permeability	Volume-averaged permeability. See Permeability.
Burnup	A measure of nuclear-reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission or as the amount energy produced per unit weight of fuel.
Burnup Credit	An approach used in criticality evaluations which accounts for a reduction in criticality potential associated with spent nuclear fuel relative to that of fresh fuel. Burnup credit also accounts for variations in criticality potential due to radioactive decay based on the time the fuel was removed from the reactor.
Casting Yard	A facility used for the production of precast concrete elements such as the segments installed in a tunnel to establish a roadway.
Cladding	The metallic outer sheath of a fuel element generally made of a zirconium alloy or stainless steel. It is intended to isolate the fuel element from the external environment.
Climatology	The science that deals with climates and their phenomena.
Closure	The final backfilling of the repository shafts and ramps.
Colloid	A suspension of finely divided particulate matter, 10–10,000 angstroms in size, or the particulate matter so suspended, in a gaseous, liquid, or solid substance. Colloid particles do not settle out of the substance rapidly, and are not readily filtered. For this project, the concern is that colloids may adsorb radionuclides, which would then be transported by groundwater.
Contact (Geology)	A plane or irregular surface between two different types or ages of rocks.
Containment	The retention, by physical barriers, of radioactive waste within a designated boundary for thousands of years.
Controlled Area	A surface location, to be marked by suitable monuments, extending horizontally no more than 10 km (6.21 miles) in any direction from the outer boundary of the underground facility, and the underlying subsurface, which area has been committed to use as a geologic repository and from which incompatible activities would be prohibited before and after permanent closure.

Criticality	The condition in which nuclear fuel sustains a nuclear chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. For the TSPA-VA, it is a condition that would require the original waste form, which is part of the waste package, to be exposed to degradation followed by conditions that would allow concentration of sufficient nuclear fuel, the presence of neutron moderators, the absence of neutron absorbers, and favorable geometry.
Critical Configuration	Arrangement of fissionable material for which the neutron multiplication factor is equal to one, resulting in a self-sustaining chain reaction.
Cross Drift	An excavation across the block of the proposed repository excavated in a general east-west direction.
Defense in Depth	The term used to describe the property of a system of multiple barriers to mitigate uncertainties in conditions, processes, and events by employing barriers that are redundant and independent, such that failure in any one barrier does not result in failure of the entire system.
Design Alternative	A considered alternative to a major design feature that is important to waste isolation. 10 CFR 60.21(c)(1)(ii)(D) requires that the safety analysis report include a comparative evaluation of the major design features, important to waste isolation, that have been considered with particular attention to the alternatives that would provide longer radionuclide containment and isolation.
Design Basis Events	Naturally or humanly induced events that may occur before permanent closure of the geologic repository's operations area.
Design Margin	Margins of safety in specifications for engineered components to account for uncertainty in the conditions to which the components will be subjected and for variability in the properties of component materials.
Design Option	An engineered barrier system feature being considered for possible inclusion in the LA design. The three design options discussed in the VA reference design are waste package ceramic coating, waste package drip shields, and emplacement drift backfill.
Desorption	A physical or chemical process by which a substance that has been adsorbed or absorbed by a liquid or solid material is removed from the material.
Diffusion	The spreading or dissemination of a substance. The gradual mixing of the molecules of two or more substances caused by random thermal motion.
Diffusive Flow	Movement of liquid water caused by a difference in water content.
Diffusive Transport	Movement of solutes because of their concentration gradient. The process in which substances carried in groundwater move through the subsurface by means of diffusion because of a concentration gradient.

Dispersion	In hydrology, the spreading or dilution of solutes in groundwater caused by mechanical mixing with better quality water during groundwater movement and by molecular diffusion (defined). Dispersion causes dilution of solutes, including radionuclides, in groundwater and is usually an important mechanism for spreading contaminants in low flow velocity situations.
Disposal	The emplacement of high-level radioactive waste, spent nuclear fuel, or other highly radioactive material in a repository with no foreseeable intent of recovery, whether or not such emplacement permits the recovery of such waste. Isolation of such waste from the accessible environment.
Disruptive Event	An unexpected event which, in the case of the repository, includes human intrusion, volcanic activity, seismic activity, and nuclear criticality. Disruptive events have two possible effects; (1) direct release of radioactivity to the surface, or (2) alteration of the expected behavior of the system.
Distribution Function	A function whose values are the probabilities that a random variable assumes a value less than or equal to a specified value. Synonymous with cumulative distribution.
Drift	From mining terminology, a horizontal underground passage. The nearly horizontal underground passageways from the shaft(s) to the alcoves and room(s). Includes excavations for emplacement (emplacement drifts) and access (access mains).
Drill Pads	Level working areas established to support drilling operations.
Emplacement	The act of placing waste containers in prepared positions.
Engineered Barrier System	The waste packages and the underground facility. These are the designed, or engineered, components of the disposal system and the waste package.
Environmental Impact Statement	A detailed written statement to support a decision to proceed with major federal actions affecting the quality of the human environment. This is required by the National Environmental Policy Act (NEPA). The EIS describes: "...the environmental impact of the proposed action; any adverse environmental effects which cannot be avoided should the proposal be implemented; alternatives to the proposed action (although the Nuclear Waste Policy Act, as amended, precludes consideration of certain alternatives); the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented." Preparation of an EIS requires a public process that includes public meetings, reviews, and comments, as well as agency responses to the public comments. A final EIS for the Yucca Mountain site is to be published in fiscal year 2000.

Environmental Plots	Areas established for the study of environmental concerns, such as the rate of revegetation.
Expected Value	A variable's mean, or average, outcome. The weighted average of the number of possible outcomes, with each outcome being weighted by its probability of occurrence.
Exploratory Studies Facility	An underground laboratory at Yucca Mountain that includes a 7.9-km (4.9-mile) main loop (tunnel), a 2.8-km (1.75-mile) cross drift, and a research alcove system constructed for performing underground studies during site characterization. The data collected will contribute toward determining the suitability of the Yucca Mountain site as a repository, and some or all of the facility may eventually be incorporated into the proposed repository.
Fault (Geologic)	A fracture in rock along which movement of one side relative to the other has occurred.
Flow Path	The route that groundwater follows in moving from one point to another.
Flux	The volume of fluid flowing through a unit area per unit time. Also known as specific discharge.
Fracture	A general term for any break in a rock, whether or not it causes displacement, caused by mechanical failure by stress. Fractures include cracks, joints, and faults. Fractures may act as pathways for rapid groundwater movement.
Geochemistry	The study of the distribution and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere and the chemical interactions between these phases.
Geologic Exposures	Rock surfaces that are exposed when soil is removed.
Geosphere	The combination of the earth's rock, water, and air layers (spheres).
Groundwater Flux	The rate of groundwater flow through a unit area of the aquifer measured perpendicular to the direction of flow.
High-Level Radioactive Waste	(1) Radioactive material resulting from the reprocessing of spent nuclear fuel, liquid waste produced directly in reprocessing, and solid material derived from such liquid waste that contains fission products in sufficient concentrations to require permanent isolation. (2) Other highly radioactive material that NRC, consistent with existing law, determines by rule requires permanent isolation.
Hydrogeologic Properties	The properties of a rock that govern the entrance of water, and the rock's capacity to hold, transmit, and deliver water.
Hydrologic Unit	Any soil or rock unit or subsurface zone that affects the storage or movement of groundwater by its porosity or permeability.

Igneous	(1) A type of rock formed from a molten, or partially molten, material. (2) An activity related to the formation and movement of molten rock either in the subsurface (plutonic) or on the surface (volcanic).
Infiltration	The flow of a fluid into a substance through pores or small openings. It connotes flow into a substance, as opposed to percolation, which connotes flow through a substance. The process of water entering the soil at the ground surface.
In Situ	In its natural position or place. The phrase distinguishes in-place experiments, conducted in the field or underground facility, from those conducted in the laboratory.
In Situ Pore Liquid Water	Subsurface water contained in the spaces of pores in rock or soil.
Invert	(1) The low point of something such as a tunnel, drift, or drainage channel. (2) An engineered structure or material placed on excavated drift floors (the low points) to serve as structural support for drift transportation or emplacement systems. (For precast concrete, the proper name is invert segments, but they are commonly referred to simply as inverts.) Typical invert (segments) convert rounded excavated floors to flat level surfaces for transportation system use. Emplacement drift inverts may be specially designed to enhance the waste isolation and criticality prevention capabilities of the proposed repository through choice of invert materials or invert shape. Inverts may also be used to help channel water to improve repository drainage.
Isolation	Inhibiting the transport of radioactive material so that the amounts and concentrations of this material entering the accessible environment will be kept within prescribed limits.
Issue	A question relating to the performance of the monitored geologic repository that must be resolved to demonstrate compliance with the applicable federal regulations (including 10 CFR 60, 10 CFR 960, applicable EPA standards, and 10 CFR 20).
Issue Resolution	Typically refers to a provision by DOE and its contractors, and/or assembly by NRC staff and its contractors, of sufficient information about an issue such that the NRC staff has no further questions or comments (i.e., items remaining open) at a point in time, regarding how the DOE program is addressing the issue.
Key Technical Issue	Any one of 10 issues defined by NRC, based on their perception of the relative importance of the issue to the performance of the repository.
License Application	An application to NRC for authorization to construct a repository.
Licensing Safety Case	The basis for determining that the proposed repository will perform in a safe manner as required by NRC regulation during preclosure operations and postclosure periods. The safety case is defined in Volume 4, Section 2.

Low-Level Radioactive Waste	Radioactive waste that is not classified as high-level radioactive waste, transuranic waste, or byproduct tailings containing uranium or thorium from processed ore. Usually generated by hospitals, research laboratories, and certain industries.
Mineralization	The process of developing or hastening mineral formation usually from a liquid or gaseous state.
Monitored Geologic Repository	A system, requiring licensing by NRC, intended or used for the permanent underground disposal of radioactive waste (including spent nuclear fuel). A geologic repository includes; (1) the geologic repository operations area, and (2) the geologic setting within the controlled area that provides isolation of the radioactive waste. The repository will be monitored between emplacement of the last waste package and closure.
Muck Storage Areas	Surface storage areas for rock excavated from underground tunnels.
National Environmental Policy Act	The federal statute that is the national charter for protection of the environment. The act is implemented by procedures issued by the Council on Environmental Quality and DOE. The National Environmental Policy Act of 1969 appears at 42 U.S.C. 4321 et seq.
Natural Barrier	The physical, mechanical, chemical, and hydrologic characteristics of the geologic environment that individually and collectively act to minimize or preclude radionuclide transport.
Natural System	A host rock suitable for repository construction and waste emplacement and the surrounding rock formations. It includes natural barriers that provide containment and isolation features by limiting radionuclide transport through the geohydrologic environment to the biosphere and providing conditions that will minimize the potential for human interference in the future.
Near Field	The region where the adjacent natural geohydrologic system has been significantly impacted by the excavation of the repository and the emplacement of the waste.
Near-Field Environment	The condition of the area near the repository. This condition may change due to heat, water influx, and chemical changes in the rock itself.
Near-Field Geochemical Environment	The near-field environment that is subject to alteration of geological conditions through reaction to chemicals from the repository caused by the influx of gasses, liquids, or vapors.
Net Infiltration	Infiltration minus water lost to evapotranspiration and other processes such as circulation of air within the rock mass.
Neutron Multiplication Factor	The ratio of the number of neutrons present in one neutron generation to the number of neutrons from the preceding generation. When this ratio is equal to one, the fissionable material is in a critical state. When this ratio is above one, the material is supercritical, and when below one, the material is considered subcritical.

NRC Regulations	Regulations in Title 10 of the CFR that implement NRC governing requirements for the NRC organization and its licensees as authorized by the Atomic Energy Act, as amended, and the Energy Reorganization Act of 1974.
Nuclear Waste Policy Act (42 USC 10101 Et Seq.)	The federal statute enacted in 1982 that established the OCRWM and defined its mission to develop a federal system for the management and geologic disposal of commercial spent nuclear fuel and other high-level radioactive waste, as appropriate. The act also specified other federal responsibilities for nuclear waste management, established the Nuclear Waste Fund to cover the cost of geologic disposal, authorized interim storage under certain circumstances, and defined interactions between federal agencies and the states, local governments, and Indian tribes. The act was amended in 1987 and 1992.
NUREG - Series Reports	Uniquely identified NRC guidance reports, including those prepared for international agreements that cover a variety of regulatory, technical, and administrative subjects. Brochures that usually include directories, manuals, procedural guidance and newsletters and are often intended primarily for internal use. Conference proceedings that are a compilation of papers that have been presented at a conference or workshop. Books, prepared by a contractor for NRC, which serve a unique technical purpose or an industry-wide need.
Operational (Operational) Phase	The period between receipt of the first waste package and emplacement of the last waste package in the repository that includes ongoing construction of subsurface facilities, and operation and maintenance of subsurface and surface facilities, performance confirmation activities, procurement of waste packages, and operation of the transportation system.
Oxidation	A chemical reaction, such as the rusting of iron, that increases the oxygen content of a substance. A reaction in which an element's valence is increased as a result of losing electrons.
Percolation	The passage of a liquid through a porous substance (e.g., the movement of water, under hydrostatic pressure developed naturally underground, through the interstices and pores of the rock or soils).
Percolation Flux	Volume of water moving downward and/or laterally through the unsaturated zone in a given period.
Performance Assessment	An analysis that predicts the behavior of a system or system component under a given set of constant and/or transient conditions. Performance assessments will include estimates of the effects of uncertainties in data and modeling.
Performance Confirmation	The program of tests, experiments, and analyses conducted to evaluate the accuracy and adequacy of the information used to determine with reasonable assurance that the performance objectives for the period after permanent closure will be met.

Permeability	In general terms, the capacity of a medium like rock, sediment, or soil to transmit liquid or gas. Permeability depends on the substance transmitted (e.g., oil, air, and water) and on the size and shape of the pores, joints, and fractures in the medium and the manner in which they are interconnected. "Hydraulic conductivity" has replaced "permeability" in technical discussions relating to groundwater.
Preclosure	The period before and during the closure of the geologic repository.
Qualified (Data)	Data collected or developed under an NRC-approved program (including technical reviews) or previously unqualified data that have been qualified in accordance with the quality assurance requirement document.
Radioactive Waste	High-level radioactive waste and other radioactive materials, including spent nuclear fuel, that are received for emplacement in a geologic repository.
Radiological Safety	Safety from harmful exposure to ionizing radiation. The primary concerns are the potential for radiation exposure of the general public at the boundary of the controlled area of the site and the potential for radiation exposure of project workers.
Radiologically Controlled Area	An area of the surface repository enclosed by security fences, control gates, lighting and detection systems, and which includes the facilities and transportation systems required to receive and ship rail and truck waste shipments, prepare shipping casks for handling, and load waste forms into disposal containers for underground emplacement. It also includes the facility and systems required to treat and package site-generated, low-level radioactive waste for offsite disposal.
Radiolysis	The chemical dissociation of molecules caused by radiation.
Radionuclide	Radioactive type of atom with an unstable nucleus that spontaneously decays, usually emitting ionizing radiation in the process. Radioactive elements are characterized by their atomic mass and number.
Radionuclide Transport	The movement of radionuclides from the waste package to the accessible environment.
Record of Decision	A document that provides a concise public record of a decision made by a government agency.
Repository Construction	All excavation and mining activities associated with the construction of shafts, ramps, and necessary openings in the underground facility, preparatory to radioactive-waste emplacement, as well as the construction of necessary surface facilities, but excluding site characterization activities.
Repository Horizon	The near horizontal plane within the host rock where the location of the proposed repository is planned.
Release-Rate Model	Computer software routines that are used to determine the rate of release of radionuclides as waste forms break down.

Richards' Barrier	A water diversion barrier consisting of a sloped layer of gravel below a layer of finer-grained material (usually sand) that diverts water because of the combined effect of greater water retention by the finer material and higher permeability of the gravel.
Rockbolt	A bar, usually constructed of steel, that is anchored into predrilled holes in rock as a support device. The method of ground support utilizing such devices to support wall and ceilings in underground excavations.
Safety Case	A set of arguments that show that the proposed repository system will contain and isolate waste sufficiently to protect public health and safety. Underlying this set of arguments is a documented understanding of how the repository is likely to perform.
Saturated Zone Flow and Transport	Groundwater flow and the transport of substances by groundwater in an aquifer, opposed to percolation or transport in the unsaturated zone.
Seal	An engineered barrier to prevent radionuclide migration or the intrusion of undesirable substances. Specifically, in waste package closure, sealing of the lids by welding the closure lids into place.
Seepage	The slow flow of water through the pores or interstices of a substance. Specifically the amount of percolation flux that enters the emplacement drifts in a given time period. See Percolation Flux.
Seismic	Pertaining to, characteristic of, or produced by earthquakes or earth vibrations.
Seismicity	A seismic event or activity such as an earthquake or earth tremor; seismic action.
Sensitivity Study (Analysis)	An analytic or numerical technique for examining the effects of varying specified parameters when a model run is performed; shows the effects that changes in various parameters have on model outcomes and can illustrate which parameters have a greater impact on the predicted behavior of the system being modeled. Also, called sensitivity analysis because it shows the sensitivity of the consequences (such as radionuclide release) to uncertain parameters (such as the infiltration rate that results from precipitation).
Shielding	Any material that provides radiation protection. The waste package will have limited shielding, designed only to protect the outer lining of the waste package from radiolysis.
Site	A potentially acceptable area or candidate area before the establishment of the controlled area, after which time the site and the controlled area are the same.

Site Characterization	Activities, whether in the laboratory or in the field, undertaken to establish the geologic conditions and the ranges of the parameters of a candidate site relevant to the location of a repository, including borings, surface excavations, excavations of exploratory shafts, limited subsurface lateral excavations and borings, and in situ testing needed to evaluate the suitability of a candidate site for the location of a repository, but not including preliminary borings and geophysical testing needed to assess whether site characterization should be undertaken.
Site Recommendation	A recommendation by the Secretary of Energy to the President that the Yucca Mountain site be approved for development as the nation's first high-level radioactive waste repository. If the site is determined to be suitable, this recommendation is expected in fiscal year 2001.
Sorption	The binding, on a microscopic scale, of one substance to another, and includes both adsorption and absorption. In this document, the word is especially used for the sorption of dissolved radionuclides onto aquifer solids or waste package materials by means of close-range chemical or physical forces.
Source Term	Types and amounts of radionuclides that are the source of a potential release of radioactivity from the repository.
Spent Nuclear Fuel	Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. This fuel is more highly radioactive than it was before irradiation, and it is hot. See also Burnup.
Stochastic	Involving a variable (such as temperature, porosity) which may take on values of a specified set with a certain probability; data from a stochastic process is an ordered set of observations, each of which is one item from a probability distribution; random.
Stratigraphy	The branch of geology that deals with the definition and interpretation of the rock strata, the conditions of their formation, character, arrangement, sequence, age, distribution, and especially their correlation by the use of fossils and other means of identification.
Surface Facilities	The above ground repository support facilities within the restricted area.
Systems Engineering	The application of scientific and engineering principles to control a complex total-system development effort to achieve the best balance of all system elements; a process that transforms and integrates operational needs and requirements into a description of system requirements to maintain the overall system effectiveness.
Test Program	A planned set of activities, tasks, and demonstrations used to perform laboratory, vendor, prototype, system integration, equipment acceptance, startup, mockup, operational, or developmental tests of structures, systems, and components.

Thermal-Hydrologic	Of or pertaining to changes in groundwater movement caused by changes in temperature.
Thermal Loading	The application of heat to a system, usually measured in terms of watt density. The thermal loading for a repository is the watts per acre produced by the radioactive waste in the active disposal area. The spatial density at which waste packages are emplaced within the repository as characterized by the areal power density and the areal mass loading.
Thermal Mechanical	Of or pertaining to physical movement or deformation of rock because of changes in temperature.
Thermal-Mechanical Effects	Changes in the geomechanical properties of the repository host rock produced by heating of the rock associated with the emplacement of radioactive waste in the repository. An example might be decreased rock strength related to increased fracturing caused by heating of the rock.
Thermogravimetric Analysis	A method of analysis that measures the loss or gain of weight by a substance as the temperature of the substance is raised or lowered at a constant rate.
Total System Performance Assessment	A risk assessment that quantitatively estimates how the proposed Yucca Mountain repository system will perform in the future under the influence of specific, features, events, and processes, incorporating uncertainty in the models and data.
Tracer Testing	A procedure in which a soluble substance (tracer) is added to groundwater at one location and its movement to another location is observed. Tracer testing is a technique by which groundwater flow directions and velocities and other hydrologic properties of rocks can be estimated.
Trade-Off Studies	A systematic evaluation of alternative concepts, designs, arrangements, configurations, and technologies to compare and select the preferred option.
Transparency	According to the Nuclear Waste Technical Review Board, the ease of understanding the process by which a study was carried out, which assumptions are driving the results, how they were arrived at, and the rigor of the analyses leading to the results. According to a peer review panel report, transparency "requires ensuring completeness and using a logical structure that facilitates in-depth review of the relevant issues...achieved when a reader or reviewer has a clear picture of what was done in the analysis, what the outcome was, and why."
TSPA Base Case	A reference case, based on current understanding, against which to compare variation of parameters when performing a sensitivity analysis. This may include less likely events; from regulatory language; the situation of the area being modeled without unlikely or unanticipated features, events, or processes represented implying that anticipated features, events, and processes are the foundation of the base case.
Tuff	Igneous rock formed from compacted volcanic fragments from pyroclastic (explosively ejected) flows with particles generally smaller than 4 mm in diameter. The most abundant type of rock at the Yucca Mountain site.

Unsaturated Zone	The zone of soil or rock between the land surface and the water table.
Unsaturated Zone Flow	The flow of water in the unsaturated zone by downward percolation and by capillary action.
Waste Form	A generic term that refers to radioactive materials and any encapsulating or stabilizing matrix.
Waste Package	A loaded, sealed, and tested disposal container.
Waste Package Closure	Sealing of the waste package by welding the closure lids into place.
Waste Package Degradation	Reduction in the integrity of waste package while in the repository. This occurs over time and could result from corrosion and other processes such as oxidation and thermal embrittlement of the container material(s).
Work Breakdown Structure (WBS) Index and Dictionary	In use for the YMP and consisting of two parts: an index and a dictionary. Part 1, the index, lists the WBS elements in an indented format to show their hierarchical relationship. Part 2, the dictionary, defines the objective and the description of work for each individual WBS element.
WBS Elements	Unique work scope components that are identified to facilitate effective project management.

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Viability Assessment of a Repository at Yucca Mountain