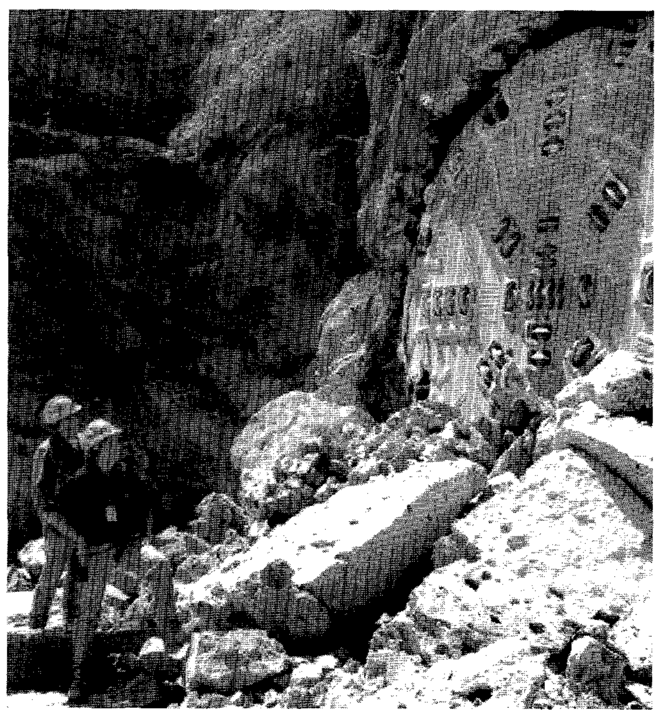
DOE/RW-0505

Site Characterization Progress Report
Yucca Mountain, Nevada



Nuclear Waste Policy Act
(Section 113)

April 1, 1997 - September 30, 1997
Number 17
April 1998

U.S. Department of Energy
Office of Civilian
Radioactive Waste Management

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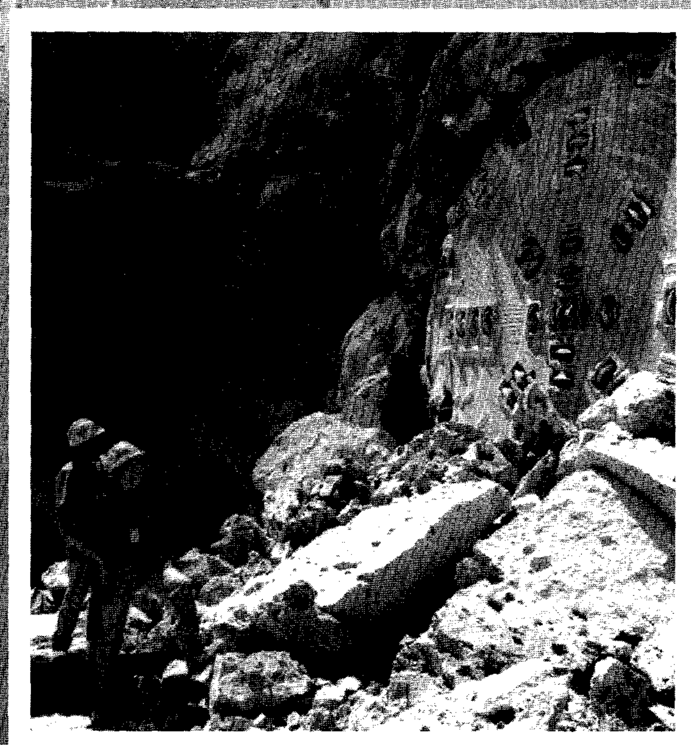

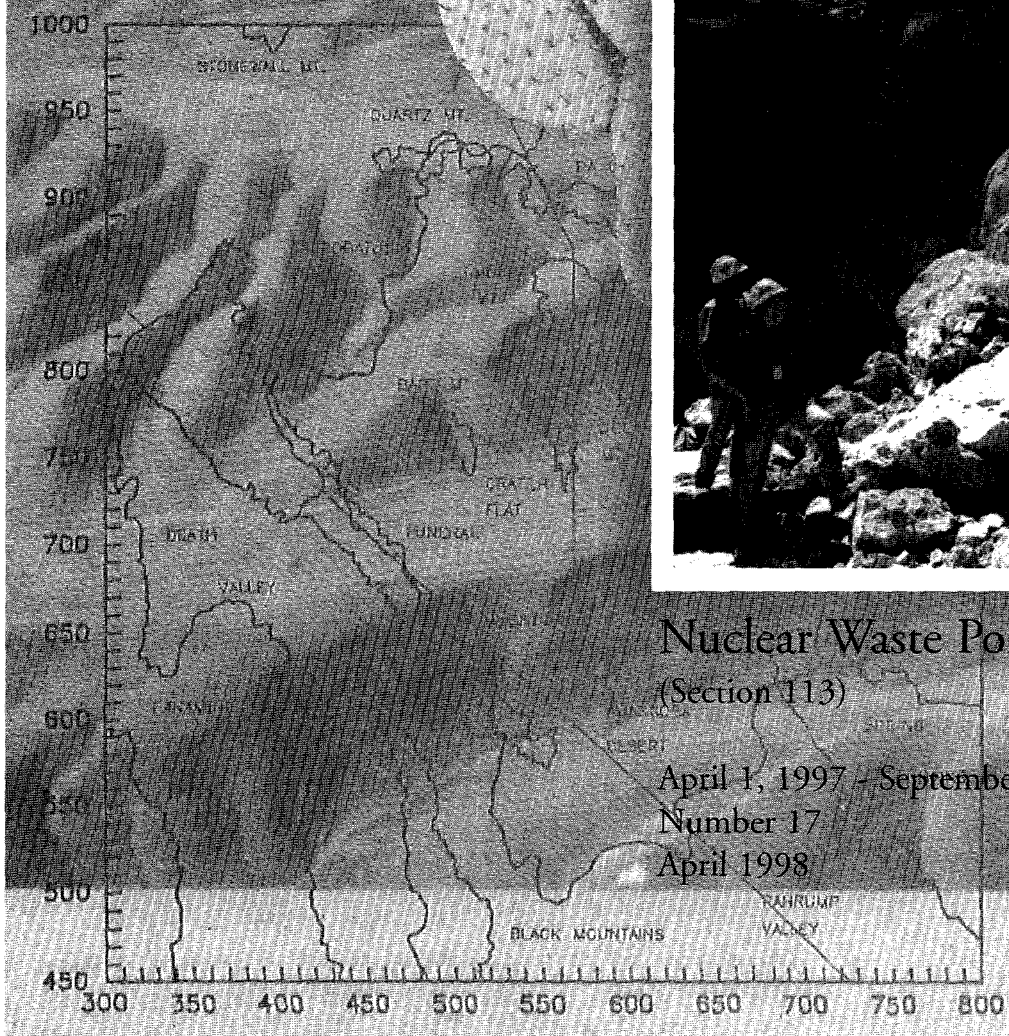

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MASTER

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Department of Energy
Washington, DC 20585

June 10, 1998

Note to Readers

Progress

During the six-month reporting period from April 1 to September 30, 1997, we continued implementing the revised program approach outlined in the 1996 Program Plan. The program is currently focused on completing the work for the 1998 viability assessment that represents the culmination of more than 15 years of site characterization activities and progress in designing and evaluating a repository system.

The viability assessment will provide a basis for making an informed assessment of the feasibility to proceed with the process of licensing and constructing a repository at Yucca Mountain. From the Program's perspective, the viability assessment components will objectively describe the design, performance, and cost of a Yucca Mountain repository based on information collected to date. The assessment will also include a path forward for completing site characterization and developing a site recommendation and license application. The viability assessment involves consolidating the results of many years of work into documentation that includes: (1) a total system performance assessment that synthesizes scientific, design and engineering information to predict the repository's probable performance under a range of conditions and various design options, over thousands of years; (2) preliminary design concepts for the repository and waste package; (3) a plan for developing a license application and an estimate of what executing that plan would cost; and (4) an estimate of the costs to construct, operate, and close a repository, based on the preliminary design concepts.

To support the total system performance assessment, 10 abstraction and testing workshops were conducted, including nuclear criticality, saturated-zone flow and transport, and biosphere process models. The workshops represented efforts to simplify the corresponding individual process models to determine the relevant influences on total system performance, define each model's most significant uncertainties, and determine methods to address those uncertainties. To further strengthen total system performance assessment tools, three formal expert elicitations were conducted on unsaturated-zone flow, waste package degradation, and saturated-zone flow and transport. The experts addressed models, parameters, issues, and ranges of uncertainty that will be used in sensitivity analyses. Together, the results from the abstraction and testing workshops and sensitivity analyses will improve the models and refine the total system performance assessment.

Other site investigations supporting performance assessment include the drift scale heater test that began on December 3, 1997, in the Thermal Testing Facility; the large block test at Fran Ridge; tracer tests and long-term hydraulic tests at the C-well complex; and studies of the relationship between percolation flux and seepage into drifts in the Exploratory Studies Facility. Additional activities underway to support testing are construction of the cross drift launch chamber for the tunnel boring machine and construction of the Busted Butte facilities for geochemistry tests of radionuclide transport through the tuff.

Design activities continued to focus on the design of first-of-a-kind repository systems, structures, and components as well as addressing design issues that will have major impacts on performance assessment, construction options, and repository cost and schedule. The repository and waste package reference designs were provided at the end of the reporting period. Refinements to these designs are expected prior to their use as inputs to total system performance assessment, repository system design, and cost estimates supporting the viability assessment. The four design options under evaluation as potential benefits to repository performance include: (1) taking credit for the fuel cladding; (2) use of backfill; (3) a waste package drip shield; and (4) a ceramic waste package coating. The evaluations are quantitative but include significant assumptions; these evaluations should be completed by mid-fiscal year 1998.

Format

The Progress Report format has been revised to provide a concise, summary-level document and improve readability and usefulness. The new reporting format includes an introduction and executive summary, a discussion of work planned to address near-term objectives, and a description of progress in repository performance, design and construction, and site characterization. Progress Report 17 represents a departure from the format of the 1988 Site Characterization Plan, but provides a more effective and well integrated view of Yucca Mountain Site Characterization Project progress for the reporting period.

Progress Report 18 will be a letter report and will be issued in July 1998. Our decision to streamline the report length will allow us to concentrate efforts on our primary focus, the 1998 viability assessment. In Progress Report 18, summaries will emphasize site characterization and design progress and briefly discuss progress on the four components of the viability assessment.

Progress Report 19, covering the period April 1 through September 30, 1998, is expected to be issued in March 1999.

In previous Progress Reports, Appendix A documented the history of changes in site characterization, repository and waste package design, and performance assessment from those described in the 1988 Site Characterization Plan. With the issuance of Progress Report 17, this information has been presented in a separate document, Documentation of Program Change (YMP/98-03). During the remainder of the site characterization process, the Documentation of Program Change will be revised annually, reflecting changes and updates to the Project that occurred during the fiscal year.

Sincerely

Lake H. Barrett, Acting Director
Office of Civilian Radioactive
Waste Management

SITE CHARACTERIZATION PROGRESS REPORT: YUCCA MOUNTAIN, NEVADA

April 1, 1997 to September 30, 1997

Number 17

PROGRESS REPORT #17

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CHAPTER 1 – INTRODUCTION AND SUMMARY

INTRODUCTION

The U.S. Department of Energy's (DOE) Office of Civilian Radioactive Waste Management (OCRWM), created with the enactment of the Nuclear Waste Policy Act of 1982 (NWPA), is tasked to accept and dispose of the nation's high-level radioactive waste and spent nuclear fuel in a deep geologic repository (high-level radioactive waste program). The OCRWM provides overall management for the high-level radioactive waste program. Within the OCRWM program, two project offices, the Office of Waste Acceptance, Storage, and Transportation and the Yucca Mountain Site Characterization Office, operate to accomplish the ultimate goal of disposal. This Progress Report presents the progress of the Yucca Mountain Site Characterization Project (Project) where scientific study, performance, and design activities are being conducted for development of a potential repository.

This is the seventeenth semiannual report of the Yucca Mountain Site Characterization Project. The report summarizes significant site characterization activities during the period from April 1, 1997 through September 30, 1997, in the evaluation of Yucca Mountain as a potential site for the geologic disposal of spent nuclear fuel and high-level radioactive wastes. The location of the Yucca Mountain site is depicted on Figure 1-1. The progress report also cites technical reports and research products that provide the detailed information on these activities.

A report on site characterization progress is required semiannually by the Nuclear Waste Policy Act of 1982, as amended (NWPA), Section 113(b)(3), and 10 CFR 60.18(g). The report must describe the progress of site characterization activities and the information developed, as well as waste form and package research and development; identify new issues and plans to resolve these issues; discuss the elimination of planned studies no longer necessary to site characterization; and identify decision points reached and schedule modification.

This chapter provides an introduction to the Report and presents an Executive Summary of Project progress toward achieving the goals of the 1996 *Civilian Radioactive Waste Management Program Plan* (Program Plan) (DOE 1996a). The chapter includes an explanation of the revised format used in this progress report, a brief history of the Project, a description of the programmatic and technical framework that guides Project work, a summary of Project progress for the reporting period, and items of importance that have occurred after the reporting period. Section 1.4 serves as an Executive Summary of this semi-annual progress report. An epilogue is provided in Section 1.5, listing Project accomplishments since the end of the reporting period.

Chapter 2 outlines technical and regulatory issues that must be addressed by the Project and planned work toward achieving future objectives concerning the viability assessment, the environmental impact statement, the site recommendation, and the license application. Chapter 3 describes technical progress in preclosure radiological safety analysis, postclosure performance assessment, and performance confirmation activities. Chapter 4 describes various aspects of repository and waste package design and construction. It also discusses the Exploratory Studies

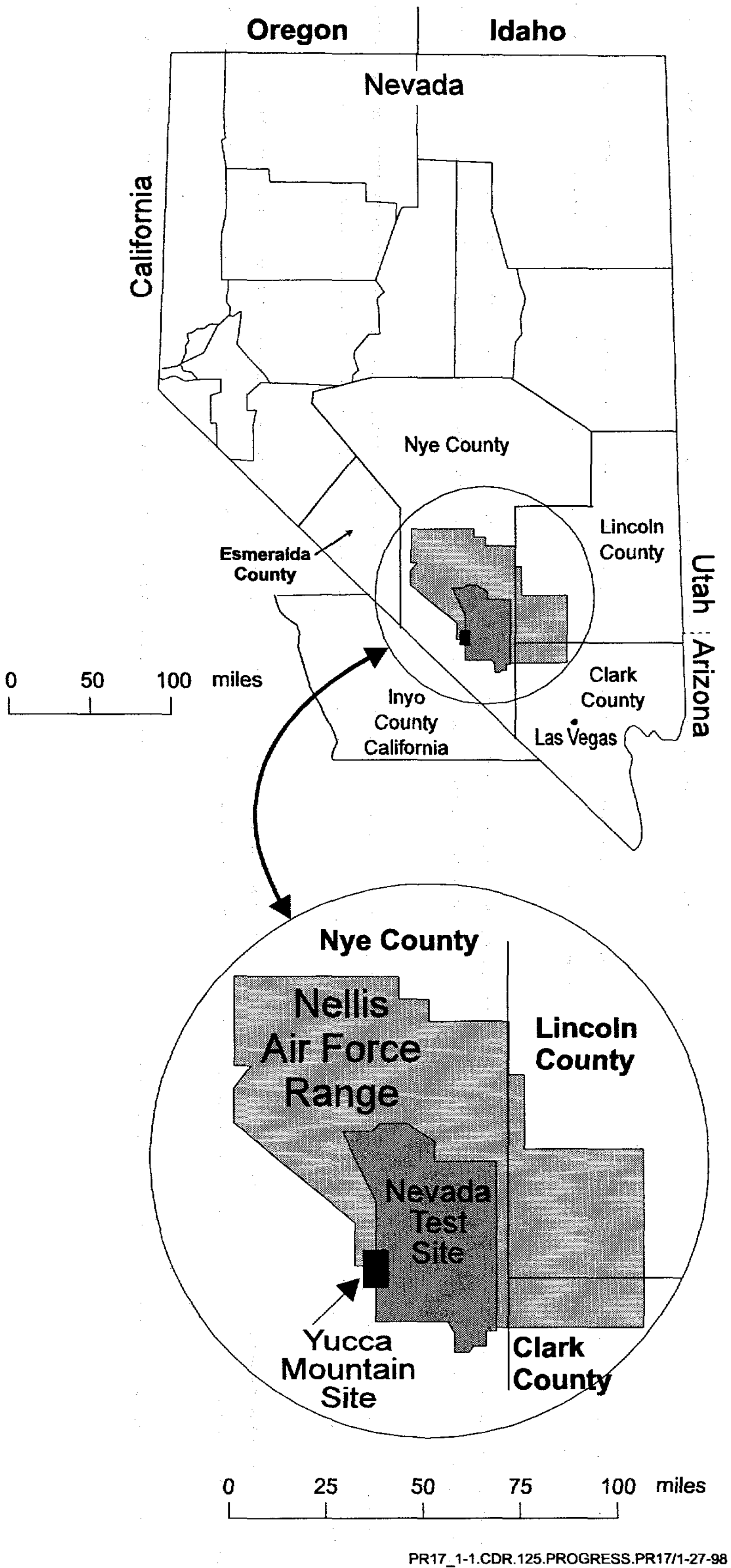

Figure 1-1. Location of the Yucca Mountain Site

Facility cross drift. Chapter 5 describes site characterization activities, and Chapter 6 contains a complete list of references.

An associated document, *Documentation of Program Change* (CRWMS M&O 1998), provides a systematic review and documents changes in the site characterization program since it was first delineated in the *Site Characterization Plan, Yucca Mountain Site Research and Development Area, Nevada* (Site Characterization Plan) (DOE 1988), thus enabling the reader to determine the present scope of that program and how it has changed since 1988. This document previously was Appendix A to Progress Reports #14, #15, and #16 (DOE 1996d; DOE 1997d; DOE 1997b).

1.1 FORMAT

The format of this Progress Report has been revised to provide readers with a more concise, summary-level document that focuses on the major, near-term objectives of the Project, highlights significant progress in achieving those objectives, and presents an integrated view of Project activities. The new format is designed to increase the readability and usefulness of the document and the timeliness in which it is prepared and distributed. As a result, the reader should find it easier to understand the site characterization process, and the Project should be better able to outline long-term strategies and goals, and more effectively describe Program and Project activities and plans.

This report will continue to comply with statutory and regulatory requirements and will focus on current significant technical progress in repository performance, design, and site investigation activities, as well as progress to achieve programmatic objectives. In previous progress reports, the technical discussions were arranged by Site Characterization Plan (SCP) activity or task identifier. The technical discussions in this Progress Report have now been consolidated and are presented as summaries of broader topic areas. Detailed technical information will no longer be listed directly in the text, but will be accessible through the references.

Short citations in the text will be used for completed and “in press” or “in prep.” references. However, the full citations will be listed separately at the end of this document. The “in prep.” references are currently in development or in the publication process and are subject to revision before completed; when completed, these documents may be obtained from the Project.

1.2 PROJECT HISTORY

In 1982, Congress passed the Nuclear Waste Policy Act of 1982 (NWPA), which established the Office of Civilian Radioactive Waste Management within the U.S. Department of Energy (DOE) and affirmed geologic disposal as the Nation’s long-term strategy to deal with spent nuclear fuel and high-level radioactive waste. The primary goals of the Act were to: (1) site, construct, and operate a mined geologic repository; (2) establish federal responsibility for disposal of spent nuclear fuel and other high-level radioactive wastes; and (3) establish a fund to ensure that the costs of performing activities related to geologic disposal are borne by the generators of these wastes. The NWPA was amended in 1987, and Congress directed the DOE to investigate only the potential site at Yucca Mountain.

The Energy Policy Act of 1992 (EPACT) (42 U.S.C. 10141 note), modified requirements in the NWPA, by directing that the EPA promulgate site-specific public health and safety standards, consistent with recommendations of the National Academy of Science, for "protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site." The EPA Yucca Mountain standards are still under development.

In FY 1994 Congress directed the DOE to refocus its efforts on completing the core scientific activities at Yucca Mountain. Congress also directed the DOE to complete excavation of the necessary parts of the Exploratory Studies Facility and the scientific tests needed to assess the performance of a repository, determine the suitability of the Yucca Mountain site, and complete conceptual designs for a repository and waste package.

The DOE issued the *Civilian Radioactive Waste Management Draft Program Plan* in May 1996; this plan has since been finalized (DOE 1996a). The Plan outlined the path to accomplish the changes directed by Congress. The Plan emphasized identification of issues crucial to an assessment of the site and was, in effect, endorsed by Congress in the Energy and Water Development Appropriations Act, 1997. The Act requires the DOE to complete and submit to the Congress a viability assessment. The purpose of the viability assessment is to provide policymakers with the information required to make an informed assessment of the feasibility to proceed to licensing and constructing a repository at Yucca Mountain, Nevada, based on current understanding of a preliminary design concept, system performance, a plan leading to license application, and cost to develop and operate a repository.

1.3 PROGRAMMATIC AND TECHNICAL FRAMEWORK FOR THE PROJECT

The goal of the Civilian Radioactive Waste Management System (CRWMS) is to license, construct, operate, and eventually close a geologic repository for spent nuclear fuel and high-level radioactive wastes, if the Secretary recommends the site for development as a repository. The Project has both a programmatic and a technical framework to guide its efforts to achieve that goal.

The programmatic framework is based on the requirements in the applicable statutes and regulations and sets specific milestones for the Project and the associated criteria. Under the framework, the Project has four near-term objectives of completing the viability assessment (Energy and Water Development Appropriations Act, 1997) in 1998, an environmental impact statement in 2000, a potential site recommendation to the President in 2001, and a license application to the U.S. Nuclear Regulatory Commission (NRC) in 2002.

The DOE Repository Safety Strategy to protect public health and safety after closure of a repository will provide the technical framework for the Project and will guide the performance, design, and site investigation activities. The strategy is expected to be issued in early 1998.

1.3.1 Programmatic Framework

The Energy and Water Development Appropriations Act, 1997, requires that the DOE complete a viability assessment by September 30, 1998. The viability assessment will identify the remaining significant technical questions regarding the Yucca Mountain site and will present information that will allow evaluation of the scientific, engineering, and financial feasibility of building and operating a geologic repository. The assessment will consist of the preliminary design concept for the critical elements for the repository and waste package, a more specific concept of repository operations, and an assessment of its performance in the geologic setting; an estimate of the remaining work needed to prepare a license application; and an updated estimate of the cost of licensing, constructing, and operating a repository of the specified design. Much of the work for the viability assessment could be used to support the preparation of any documentation accompanying a site recommendation to the President by the Secretary of Energy, if the site is found to be suitable, and preparation of the license application to be submitted to the NRC.

The NWPA requires that the Secretary of Energy submit to the President a final environmental impact statement with any site recommendation. The NRC would adopt the DOE's final environmental impact statement, to the extent practicable, in connection with the issuance of a construction authorization and subsequent license.

The final environmental impact statement will be developed under regulations that implement the requirements in the National Environmental Policy Act (NEPA) with three exceptions as required by the NWPA. The repository final environmental impact statement is not required to discuss (1) the need for a repository, (2) alternatives to geologic disposal, and (3) alternative sites (NWPA, Section 114(a)(1)(D)). The final environmental impact statement will consider the environmental impacts associated with three alternative thermal loading scenarios: high, intermediate, and low thermal loads. The final environmental impact statement is scheduled to be published in late FY 2000.

Following the Secretary's recommendation, if the President approves the site and the designation of the Yucca Mountain site as the repository site is permitted to take effect, the DOE, within 90 days of the effective date of that site designation, must submit a construction authorization application to the NRC (NWPA, Section 114(b)).

If the NRC issues a license to construct the repository, future Program activities will include repository construction, waste acceptance, transportation, packaging and emplacement of waste, performance confirmation monitoring and analysis, and eventual closure of the repository and decommissioning of the facility. At each major step, NRC approval must be obtained.

1.3.2 Technical Framework

DOE is updating its Repository Safety Strategy for assuring that public health and safety are protected after closure of a Yucca Mountain repository to reflect recent site data and the results of performance assessments. To be released early in 1998, the strategy defines the roles that the natural and engineered systems are expected to play in achieving the objectives of a potential repository system at Yucca Mountain. The objectives are: (1) near complete containment of radionuclides within the waste packages for thousands of years and (2) ensuring that annual radiation exposure (doses) to people living near the site will be acceptably low. The strategy incorporates the key assumption of the SCP that the potential repository horizon will remain unsaturated. Thus, the strategy continues to rely on the natural attributes of the unsaturated zone for primary protection by providing a setting where waste packages, assisted by other engineered barriers, are expected to contain wastes for thousands of years. As in the SCP strategy, the natural system from the drift wall to the accessible environment is expected to contribute to radiological protection by reducing the concentrations of any radionuclides released from the waste packages.

With the strategy as the framework, the DOE integrates site information, repository design, and assessment of postclosure performance to develop a safety case for the viability assessment and a subsequent license application. The strategy , in combination with the results of performance assessment, is also used to focus science and design work on the most important remaining issues related to postclosure safety. Current site information and a reference design are used to develop quantitative assessments of repository performance, that can be compared to performance measures. Repeated assessments have consistently indicated four attributes of the repository system that are crucial to meeting the objectives:

- Limited water contacting the waste packages
- Long waste package lifetime
- Slow rate of release of radionuclides from the waste form
- Reduction in radionuclide concentration during transport through engineered and natural barriers.

The first key attribute, "Limited Water Contacting the Waste Packages," includes consideration of the amount of potential seepage into repository drifts. Recent total system performance assessments indicate that limiting water contact with waste packages is the most important of the four attributes on repository performance.

How well each attribute performs depends on combinations of natural processes and engineered components – in other words, on multiple natural and engineered barriers and, thus, on "defense in depth." For example, water is limited from contacting the waste by a combination of (1) natural conditions that may act as barriers (such as arid climate, low flux in unsaturated zone, low seepage into drifts), and (2) engineered barriers (such as drift liners, air gaps).

Under the strategy, the key performance attributes are evaluated by (1) stating the remaining issues concerning each attribute in the form of testable hypotheses and then (2) developing and conducting the tests and analyses required to determine whether the evidence supports these hypotheses. Iterative evaluations among the site, design and performance teams produce an evolving picture of those site and design features that are most important to performance. This is the process that guides the development of the safety case – the set of arguments that will be made to show that the repository system will contain and isolate waste sufficiently to protect public health and safety. Underlying this set of arguments is an understanding of how the repository is likely to perform. The strategy provides the framework for developing that understanding.

1.4 EXECUTIVE SUMMARY OF PROJECT PROGRESS FOR THE REPORTING PERIOD

During the second half of FY 1997, activities at the Project continued to focus on implementing the key elements of the 1996 Program Plan (DOE 1996a) issued by the Office of Civilian Radioactive Waste Management of the DOE. These elements include (1) revising the DOE's siting guidelines, (2) supporting a viability assessment of a repository at the Yucca Mountain, Nevada, site, and (3) if the site is found suitable, completing an environmental impact statement in 2000, a site recommendation in 2001, and a license application for construction authorization in 2002.

The Project made significant progress toward the achievement of these Program Plan (DOE 1996a) elements during the reporting period – particularly toward completion of the 1998 viability assessment. Following are highlights of Project progress toward these elements in Sections 1.4.1 to 1.4.3, as well as the site characterization highlights in Section 1.4.4 that provide the technical bases for this progress.

1.4.1 Regulatory Framework Update – Program Plan Element 1

The DOE began efforts to update its siting guidelines (10 CFR 960) by publication of a Notice of Proposed Rulemaking and Public Hearing (61 FR 66157) on December 16, 1996. The proposed revision included addition of a subpart specific to the Yucca Mountain site to determine its suitability as a repository setting. Following a January 23, 1997, public hearing in Las Vegas, Nevada, the public comment period was extended twice. In this reporting period, the second extended public comment period on the proposed revision concluded May 16, 1997.

1.4.2 Viability Assessment – Program Plan Element 2

As prescribed in the FY 1997 Energy and Water Development Appropriations Act, the September 30, 1998, viability assessment must include the following four components:

- The preliminary design concept for the critical elements for the repository and waste package

- A Total System Performance Assessment, "based upon the design concept and the scientific data and analysis available by September 30, 1998, describing the probable behavior of the repository in the Yucca Mountain geological setting relative to the overall system performance standards"

- A plan and cost estimate for the remaining work required to complete a license application
- An estimate of the costs to construct and operate the repository in accordance with the design concept.

The DOE continues to perform the work necessary to complete the four components of the viability assessment.

Progress on the Project's current technical work is described in Chapter 3 Repository Performance, Chapter 4 Design and Construction, and Chapter 5 Site Characterization. This work includes progress in scientific, engineering, and field activities that constitute the basis for the viability assessment components and for future Project objectives. The viability assessment serves to focus the technical and cost information available in 1998 from such activities, past and present, and to establish the program of future activities that will constitute the basis for future Project objectives.

1.4.2.1 Repository and Waste Package Design

The primary focus of the engineering and design activities for the reporting period was on the development of a reference design to support the Total System Performance Assessment for Viability Assessment (TSPA-VA). Other key engineering and design activities included completion of the Exploratory Studies Facility main tunnel loop on April 25, 1997, when the tunnel boring machine emerged from the south portal (see Figure 1-2); development of a compliance program to evaluate NRC and industry guidance and standards for applicability to repository design and construction; tentative resolution of viability assessment design issues; and identification of four design options.

Currently, repository and waste package design activities are focusing on key design issues and design of critical systems, structures, and components that have little or no regulatory precedent but have a major impact on performance, schedule, construction, and cost. When complete, the viability assessment design effort will have evaluated the technical feasibility of the conceptual designs but will not have developed all the detail needed for licensing.

During this reporting period, the reference designs for the repository subsurface and surface, engineered barrier system, and waste package were developed to support the total system performance assessment. The repository subsurface layout is being designed to allow emplacement of at least 70,000 metric tons uranium (MTU) of radioactive waste with enough flexibility to accommodate potential changes in site conditions (10 CFR 60.133(b)) or programmatic requirements. Analyses and engineering drawings were completed to establish feasible system configurations for surface and site facilities and to provide preliminary facility layouts sufficient to support the viability assessment cost estimate. The current reference repository design includes an areal mass load of 85 MTU/acre (see Figure 1-3). The engineered barrier system includes a concrete drift liner, an initial air gap in the drift, a two-layer waste package with outer corrosion allowance and inner corrosion resistant material, in-drift emplacement of the waste packages, placement of the packages on a steel pedestal, and a

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Figure 1-2. Tunnel Boring Machine Emerging from the Exploratory Studies Facility on April 25, 1997

concrete invert at the base of the drift (see Figure 3-3). Possible engineered barrier system options were also identified. These options establish the basis for reliance on spent fuel cladding, ceramic coating for the waste package, waste package drip shields, and backfill for ceramic coating protection and corrosion deterrence. These options are under evaluation and may be pursued as part of the viability assessment.

During the past six months, several deliverables related to waste package and engineered barrier system materials have been completed for performance assessment. These deliverables include summaries of recent test data and performance predictions for the inner barrier, including effects on performance of pitting corrosion, crevice corrosion, stress corrosion, and galvanic corrosion. A very comprehensive report on corrosion effects at the outer-barrier/inner-barrier interface has recently been completed and is in review (Huang in prep.). An updated intrinsic spent fuel dissolution model based on non-equilibrium thermodynamics was provided for use in the TSPA-VA and will be incorporated into the Waste Form Characteristics Report.

1.4.2.2 Total System Performance Assessment

To provide a defensible evaluation of the repository system's performance, the Project is conducting an integrated TSPA-VA. This assessment will evaluate the range of probable behavior of the repository in the Yucca Mountain geologic setting. It will also help identify which components of the natural and engineered systems most influence performance to focus scientific and design work on those features of the repository that, if better defined and understood, could most increase confidence in repository performance.

Important concerns are that model development focus on issues most important to performance and that evaluation be conducted in a manner that enhances traceability (provides a complete and unambiguous record) and transparency (can be easily understood by the reader and reviewer). The Project completed documentation of the methods and assumptions to be used for the TSPA-VA (CRWMS M&O 1997cd).

Activities supporting TSPA-VA development centered on developing abstracted models to describe the different natural and engineered components of the total system, and on combining these components into a total system representation that will allow a projection of the repository's expected behavior. The total system performance assessment will include evaluations of alternative conceptual models, variability and uncertainty in parameters and processes, and a sufficient level of detail to maintain the sensitive aspects of each process. Updated site data and engineering data and assumptions used for model input will provide a more accurate projection of repository performance than in previous performance assessments. In conducting an assessment of postclosure performance, the system is broken into discrete components that represent reasonably separable processes. Each of the individual processes incorporated in the analyses is defined by a complex, behavioral process model. Model simplification or abstraction of relevant information is necessary for input to the total system model. During the year, a series of ten workshops (three in this reporting period being nuclear criticality, saturated-zone flow and transport, and biosphere) was convened to identify the most appropriate abstraction methods to assure that relevant information from each of the key process

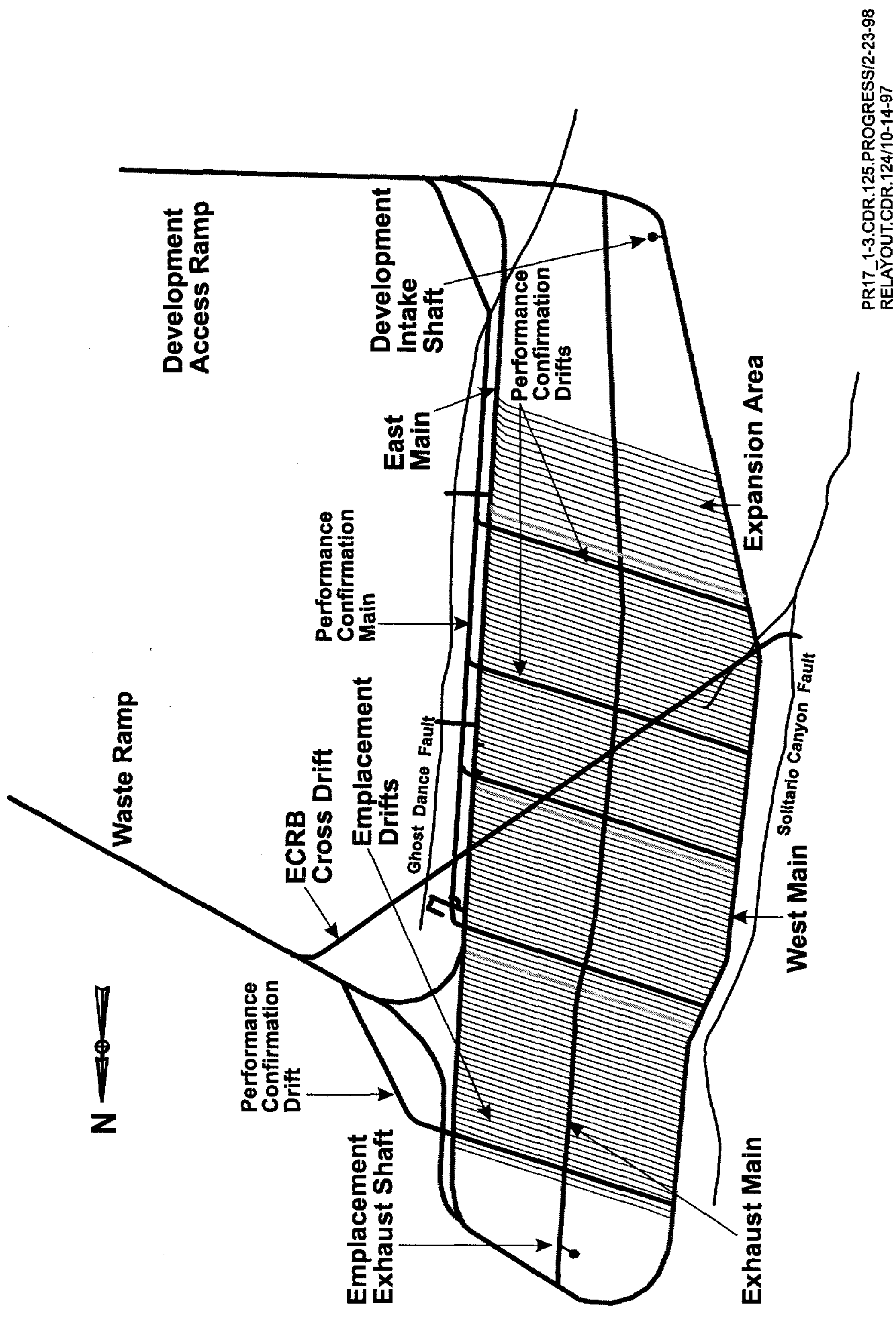

Figure 1-3. Preliminary Layout of Potential Repository

models was incorporated in the total system model. These workshops also assisted the Project in defining the most significant uncertainties associated with each process model as well as methods to address those uncertainties in the TSPA-VA. The results of these workshops and the analysis plans for the total system performance assessments were documented and implemented during this reporting period; results are expected in early FY 1998.

During this reporting period, expert elicitations were conducted on the parameters and models used to represent unsaturated-zone flow, waste package degradation, and saturated-zone flow and transport in both the process models and the performance assessment models. The members of the elicited panels were nationally recognized experts in the subject matter for each respective model.

In the unsaturated-zone flow model expert elicitation, emphasis was on estimates of surface infiltration and deep percolation. Waste package degradation experts provided corrosion parameter distributions and qualitative input on corrosion morphology, behavior, and geometry. In the saturated-zone flow and transport elicitation, multiple issues were addressed, and parameters, estimated. Examples include conceptual models of saturated-zone flow and the large hydraulic gradient, specific discharge below and down-gradient of the repository. Estimates of bounds elicited from each expert on issues specific to each model will be used to supplement the process models developed for the Project. Many of the models, parameters, issues, and uncertainty bands provided by these expert elicitations will be used in sensitivity analyses that will improve and refine the models developed in the abstraction/testing activities described above.

1.4.2.3 Plan and Cost Estimate for Completion of a License Application

The third component of the viability assessment, the license application plan, will define the remaining work, after viability assessment, required to complete a license application and the associated schedules and costs. Submittal and docketing of a license application to the NRC, should the Yucca Mountain site be found suitable, is a major objective for the DOE. During this reporting period, the DOE began development of an annotated outline of this viability assessment product.

The *Work Description for Multi-Year Project Summary Schedule* (YMP 1997g) was completed in September 1997. This product, an input to the license application plan, discussed the Project licensing strategy, license application content requirements as directed by 10 CFR 60.21, and planned Project work from viability assessment completion to completion of a license application. It also contained summary cost and schedule information and a basic description of performance confirmation activities. The current license application plan (CRWMS M&O 1997c) emphasizes determining the work needed to complete an application and will be based on the understanding of the site at viability assessment in 1998. The license application plan (CRWMS M&O 1997c) will address (1) the issues critical to overall system performance and to design, (2) the necessary site characterization, design, and performance work that will be performed between viability assessment and license application submittal, and (3) the relationship between that work and the critical issues to performance. In early FY 1998, a management plan to guide development of the license application plan (CRWMS M&O 1997c) will be completed.

1.4.2.4 Cost to Construct and Operate the Repository

The fourth component of the viability assessment includes the estimated costs and schedules to construct and operate the repository system in accordance with the critical elements of the design concept. The estimates will encompass the period following license application submittal and will include costs for performance confirmation; and construction, operation, and closure of a repository. The cost and schedule estimates will be a factor in policy decisions regarding the feasibility of, and justification for, continuing with the work required for licensing and construction of a repository.

Based on work performed in the previous reporting period, the DOE completed a cost analysis report (CRWMS M&O 1997a) that presented the assumptions and format envisioned for the viability assessment cost estimate. It provided a Project cost estimate based on the current reference design. The cost analysis report will be used to prepare the Mined Geologic Disposal System (MGDS) viability assessment cost estimate during FY 1998.

1.4.3 License Application – Program Plan Element 3

The third Program Plan element, submitting the license application to seek construction authorization, requires three important steps. All are required statutorily by the NWPA:

- A final environmental impact statement, which will be prepared to requirements in the DOE's 10 CFR 1021, *National Environmental Policy Act Implementing Procedures*
- A site recommendation to the President based on the requirements of the *Nuclear Waste Policy Act*, Section 114(a).
- Development of a license application requesting construction authorization per the NRC's requirements, which are currently set forth in 10 CFR 60, *Disposal of High-Level Radioactive Wastes in Geologic Repositories*, and submittal to the NRC.

Following completion of the viability assessment at the close of FY 1998, the DOE's focus will shift toward completing the necessary steps leading to submittal of a license application. The DOE will evaluate the suitability of Yucca Mountain as a repository site against the criteria in the then-existing siting guidelines. If the site is found suitable, the Secretary of Energy can issue a site recommendation to the President in 2001, following public hearings to be held in the State of Nevada.

A final environmental impact statement, currently planned for completion in 2000, must accompany the site recommendation to the President. If the site is approved by the President and that designation is permitted to take effect, the Secretary of Energy will submit an application for construction authorization to the NRC in 2002. This schedule would allow waste emplacement operations to begin in 2010.

Activities supporting development of the preliminary draft environmental impact statement continued during this reporting period. The environmental impact statement comment summary document (DOE 1997c) was issued in May 1997. This document summarizes comments and issues identified during the 1995 public scoping process and describes the DOE's approach to address these issues in the environmental impact statement. Total system performance assessment model abstractions and testing specific to the draft environmental impact statement were conducted. Also, scientific and engineering support activities included providing data for the environmental description for input to the environmental impact statement and conducting studies on transportation issues, options of thermal loading, retrieval, backfill, and lag storage.

During this reporting period, planning was completed for FY 1998 work in support of site suitability and site recommendation. Of particular interest, the planning activity was completed for the East-West cross drift. The approved work scope resulting from this planning activity included the design and construction of the East-West cross drift, drilling a borehole (SD-13) to the water table north of the proposed repository block, drilling a borehole (SD-11) to the water table in the southwestern part of the proposed repository block, and performing the testing associated with each of these elements. Design and construction planning for the East-West cross drift began in late FY 1997 and will continue into FY 1998. Construction of the East-West cross drift is scheduled to begin in the first quarter of FY 1998 with the start of the launch chamber. The East-West cross drift construction will be completed in September 1998, and associated testing alcoves will be completed in the second quarter of FY 1999. The two boreholes are scheduled to be drilled in FY 1999; the site data gathered from these elements will be provided as input to future performance assessment modeling that will support both the site recommendation and the license application. Also planned for FY 1998 are the development of two management plans necessary to determine compliance with 10 CFR 960 and to compile the necessary documentation, guide technical work, and put in place the processes to reach site recommendation.

In support of licensing, the DOE continued issue resolution activities with the NRC staff. A revision to the *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997b), which will serve as the basis for a topical report on the same subject to be developed in FY 1998, was developed for early FY 1998 submittal to the NRC. Progress in the series of three seismic hazard topical reports continued. Revisions to the first and second seismic topical reports, *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain* (YMP 1997a) and *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b) were completed and issued in August 1997. An annotated outline for a third and final seismic topical report, *Determination of Preclosure Seismic Design Basis for A Geologic Repository at Yucca Mountain*, was in preparation, with the review draft scheduled for completion by early FY 1998.

A major step toward license application will be to develop a working draft of the license application in 1999 as discussed in Section 2.4. This working draft will assist the Project in organizing information necessary to support eventual, successful docketing of the license application. The working draft will also help identify topics in the licensing case that will need additional attention and emphasis prior to completion of the license application for submittal to the NRC. After the working draft is completed, work will continue on development of the version of the license application that will be submitted to the NRC in 2002 to seek construction authorization.

The Project Integrated Safety Assessment, previously planned in support of the viability assessment, will not be completed, with the exception of the chapters covering performance assessment and site description. These chapters will be completed as stand-alone documents in support of the viability assessment. The topics that were to be documented in the Project Integrated Safety Assessment will now be developed in the working draft license application.

Two important documents that will support license application development were prepared during this reporting period. These documents included the *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain* (YMP 1997e) and the *Technical Guidance Document for License Application Preparation* (YMP in prep. [a]). These will provide the management and technical frameworks, respectively, for development of the license application.

1.4.4 Site Characterization Support to Program Plan Elements

The Project made important progress in the characterization of the site. Results of synthesis and modeling activities support the viability assessment and represent important advances in the development of qualified data and models to support the license application. The significant findings that can potentially improve system performance involve heat, percolation, and sorption.

Within the altered zone, expected heat-induced changes could increase sorption of radionuclides and increase groundwater transport times. Waste package life appears to be sensitive to water vapor because relative humidity appears to have a significant effect on the corrosion rate of the waste package's outer barrier. Preliminary results of moisture balance calculations showed daily losses from evaporation are the same order of magnitude as the annual percolation rate, and researchers see potential for the drying of the rocks during construction and operation because of ventilation effects. Results of laboratory studies indicate that radionuclide transport times increase as the degree of saturation in the Yucca Mountain tuffs decreases. This result suggests that drying of the repository could slow radionuclide migration rates and improve overall performance of the potential repository. Although bomb-pulse environmental isotopes, which indicate potential fast pathways, were found in the northern part of the ESF, no such evidence was found in the southern part of the ESF.

Chapter 5 describes the key advances made in site characterization.

1.5 EPILOGUE

Since the end of the reporting period of April 1, 1997, to September 30, 1997, but during production of this progress report, the following noteworthy items have occurred:

- The DOE evaluated the disposal of plutonium and in December 1997 approved a baseline change to incorporate plutonium disposal into the high-level radioactive waste program. The DOE will include plutonium disposal in documentation for the license application. Plutonium will be accepted in two forms: mixed-oxide fuel and ceramic-immobilized plutonium.

- In January 1998, the DOE released the Repository Safety Strategy (YMP 1998).

- The Drift Scale Heater Test in the Thermal Testing Facility began December 3, 1997. The largest heater test at Yucca Mountain, this is also the world's largest underground thermal test. Electric heaters will heat more than 13,000 cubic yards of rock, and scientists will study thermal, mechanical, hydrologic, and chemical processes.

- Scientists are studying the differences in water percolation flux and seepage. Tests measure water injected into boreholes above a niche, comparing rates to evaluate the bounds of flux that trigger seepage into drifts.

- The launch chamber for the East-West cross drift tunnel boring machine is being completed.

- Construction of the Busted Butte facilities is being completed for the purpose of conducting geochemistry tests of radionuclide transport through the unsaturated tuff. Approximately 90 meters of underground workings have been excavated. All Phase I and Phase II boreholes have been drilled and surveyed. Phase I testing is expected to be initiated March 1998.

- Refinement of the viability assessment reference design completed September 30, 1997, has continued through an interactive process shared by the design and performance assessment organizations. These refinements should be complete by late March 1998. This refined design will be used in TSPA-VA and will be the basis for the cost estimate component of the viability assessment.

- Corrosion testing is providing new data on the nature and extent of specimen corrosion after one year of testing.

CHAPTER 2 – PLANNED WORK TOWARD NEAR-TERM OBJECTIVES

This chapter, a new addition to the Progress Report, describes planned work in performance assessment, design, and site characterization to meet the four statutorily-required, near-term objectives as introduced in Sections 1.4.2 and 1.4.3: the viability assessment, environmental impact statement, site recommendation, and license application submittal for construction authorization. The results of the work accomplished to meet near-term objectives will also be applied, as appropriate, to later objectives. The schedules for these objectives are shown in Figure 2-1. As each objective is achieved, this chapter will evolve to reflect newer or updated objectives in the out years as appropriate.

This chapter includes a description of technical and regulatory issues with Project-wide impact and a generalized summary of planned work to accomplish the four near-term Project objectives.

2.1 TECHNICAL AND REGULATORY ISSUES

This section describes significant regulatory issues that affect the Project's progress toward its objectives. Uncertainties in the regulatory framework and other issues must be resolved to avoid delay in the licensing process. Additionally, the NRC's Key Technical Issues must be addressed. These issues are discussed in turn in this section.

EPA Standard and NRC Regulations – The NWPA, as originally enacted, directed the EPA to promulgate generally applicable standards for protection of the general environment from offsite releases of radioactive material from high-level radioactive waste repositories. In 1985 the EPA issued these standards in the form of the final rule *Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes* (40 CFR 191).

Section 801(a) (1) of the Energy Policy Act of 1992 (42 U.S.C. 10141 note) (EPACT), which amends the NWPA, directs the EPA to promulgate public health and safety standards for protection of the public from releases from radioactive materials stored or disposed of in a repository at Yucca Mountain, consistent with the recommendations of the National Academy of Sciences. Further, the NRC is directed at Section 801(b) to modify its technical requirements and criteria currently in 10 CFR 60 to be consistent with the EPA standards.

The National Academy of Sciences formed the Committee on Technical Bases for Yucca Mountain Standards to prepare its recommendations to the EPA. In August 1995, the National Academy of Sciences published its report *Technical Bases for Yucca Mountain Standards* (NAS 1995) with its recommendations to the EPA.

Based on the National Academy of Sciences report (NAS 1995) and the comments received on that report, the EPA is in the process of promulgating a public health and safety standard for Yucca Mountain. The new standard will be contained in regulation 40 CFR 197. After the EPA standard is promulgated, the NRC will revise its regulations to maintain regulatory consistency.

Activity ID	% Comp	Early Start	Early Finish	MILE	FY96 FY97 FY98 FY99 FY00 FY01 FY02
1 VIABILITY ASSESSMENT					
M1AG	100		27AUG97A	M1	YMP03M1-Appv TSPA-VA Methodology & Assump Rpt
M1AM	100		15SEP97A	M1	YMP02M1-Completion of Phase I Design for VA
M1AT	0		04SEP98*	M1	SR400M1-OCRWM Accepts Viability Assessment
2 LICENSE APPLICATION					
M0AM	0		01MAR02*	M0	R5182M0-Submit License Application to NRC
M1BC	0		01MAR02	M1	R5181M1-Complete License Application
3 EIS, ENVIRONMENT, SAFETY AND HEALTH					
M1AX	0		30JUL99*	M1	R5151M1-Publish Draft EIS
M0AA	0		25AUG00*	M0	NE1043M0-Publish FEIS
M0AE	0		29SEP00*	M0	R5171M0-Publish FEIS Record of Decision
6 SITE RECOMMENDATION					
M1AC	100		16DEC96A	M1	SLNOPR1M1-Publish Notice of Prpsd 10CFR960 Rule
M1AP	0		30JUN98*	M1	SLNOPR2M1-Publish 10CFR960 Final Rule
M1BA	0		21AUG00*	M1	R5230M1-RW-1 Rcmnds Sec. Pub. Notice of Consid.
M0AC	0		21SEP00*	M0	R5210M0-Sec. Publ. Notice of Consideration
M0AG	0		27JUN01*	M0	R5220M0-Secretary Notifies State of Site Select
M0AJ	0		27JUL01*	M0	R5200M0-Secretary Issue SR to President

Project Start	01OCT95	Early Bar	LRPB
Project Finish	29SEP10	Progress Bar	
Data Date	01OCT97	Critical Activity	
Run Date	30JAN98		

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YUCCA MOUNTAIN PROJECT
PROGRAM MILESTONES
FY98 MULTI YEAR PLAN

Sheet 1 of 1

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Date	Revision	Checked	Approved

Figure 2-1. Schedule of Program Objectives

The DOE is concerned that EPA's standard may require an unnecessarily stringent level-of-protection coupled with an unreasonable compliance location and may not be implementable.

10 CFR 960 DOE Siting Guidelines – In addition to the NRC licensing requirements currently in 10 CFR 60, the Project will also address the DOE siting guidelines at 10 CFR 960. These guidelines constitute the "general guidelines for the recommendation of sites of repositories" that DOE is required to issue under NWPAA Section 112(a). The DOE has proposed revisions to these guidelines; Section 2.3.2 discusses these revisions in greater detail.

Key Technical Issues –The NRC has defined certain Key Technical Issues that it believes must be resolved to support licensing. The NRC plans to publish Issue Resolution Status Reports on each of the ten Key Technical Issues to provide NRC perspective on progress toward resolution of each Issue. The first of these status reports was issued this reporting period on *Issue Resolutions on Methods to Evaluate Climate Change and Associated Effects at Yucca Mountain* (NRC 1997). It addressed the issue of future climate change and its effects on hydrology; this issue is a subissue of the Key Technical Issue, "Unsaturated and Saturated Flow Under Isothermal Conditions." The status report indicated that the NRC sees no open items solely related to climate issues. The DOE will respond to this status report in early FY 1998 in a letter. The letter will acknowledge receipt of the status report and clarify a few points. Additional status reports will be published during the next reporting period. The information provided in the status reports, which frequently includes acceptance criteria for resolution of the issues, will be appropriately reflected in Project planning and guidance documents such as the license application plan (discussed in Section 2.2.3) and the *Technical Guidance Document for License Application Preparation* (YMP in prep. [a]) (discussed in Section 2.4).

2.2 VIABILITY ASSESSMENT

As discussed in the next three chapters of this progress report, the Project focus during this reporting period was primarily in direct support of the viability assessment that will be submitted to the U.S. Congress in 1998. The viability assessment, mandated by Congress, will guide the completion of work required for evaluation of site suitability and preparation of a license application. The information produced for the viability assessment will better aid policy makers in reaching a determination of the prospects for recommendation of Yucca Mountain for repository licensing, construction, and authorization. The viability assessment, as well as the results of studies performed in conjunction with excavation of the Exploratory Studies Facility (ESF) will allow the DOE to refine the site characterization program first defined in the Site Characterization Plan (DOE 1988). Changes in the Site Characterization Plan since 1988, up to the present, are presented and discussed in an associated document, *Documentation of Program Change* (CRWMS M&O 1998). Much of this work will also support future planned products needed for the environmental impact statement, the site recommendation, and development of a license application. The four viability assessment components and their primary workscopes for FY 1998 supporting the viability assessment are discussed in the following subsections.

2.2.1 Mined Geologic Disposal System (MGDS) Design Concept

In FY 1998, the repository subsurface and surface, engineered barrier system, and waste package designs will be refined for viability assessment from the reference designs completed at the end of this reporting period and used as input to the TSPA-VA. Evaluation of four optional engineered barrier system features are planned to be continued for possible inclusion in the viability assessment design. The four options include establishing the basis to rely on spent fuel cladding, a waste package ceramic coating, various waste package drip shields, and backfill. The evaluation is intended to assess preliminarily performance of the options and to understand if corrosion of the waste package is reduced by the options.

2.2.2 Total System Performance Assessment for Viability Assessment

The TSPA-VA development in FY 1998 will be divided into two major elements: (1) that associated with a base case, which will be completed on January 30, 1998, and (2) a follow-up effort called sensitivity analyses. The base case will be built using the reference design and a stochastic representation of the expected, or nominal features, events, and processes. Then, additional sensitivity analyses will be performed to explore the effects of widening certain distribution ranges, the inclusion of optional design features (such as backfill and drip shields), and the effects of disturbances to the system (including volcanism, seismicity, and nuclear criticality). After the sensitivity analyses have been completed and their possible effects have been assessed, the base case will be revised as determined appropriate to include important components indicated by these sensitivity studies, and the total system performance assessment calculations will be run in their final form for the viability assessment.

2.2.3 Plan and Cost Estimate for Completion of a License Application

In early FY 1998, a management plan will be developed to guide preparation of the license application plan for viability assessment. The focus of the license application plan is to identify work needed to support completion of a license application based on the current understanding of the site at the time of viability assessment, and based on issues identified by the DOE as crucial to overall system performance. The license application plan will present the planned work between viability assessment and submittal of a license application, discuss the relationship between the work and the most significant issues, and include estimated costs and schedules.

During this reporting period, the *Work Description for Multi-Year Project Summary Schedule* (YMP 1997g) was completed as an initial step toward completion of the license application plan. This compilation focused mostly on work needed to address 10 CFR 60.21 content requirements for the license application. In FY 1998, additional work required to address performance and other issues for the license application will be described and included in the license application plan to be completed before the end of the fiscal year.

2.2.4 Cost to Construct and Operate the Repository

The cost estimate will cover the period beginning with submittal of a license application and will reflect the cost to maintain a performance confirmation program, to develop the repository and engineered barrier designs, to construct and operate the repository, and eventually to close the repository and decommission the surface facilities. The MGDS viability assessment cost estimate will be prepared during FY 1998, based on a cost analysis report (CRWMS M&O 1997a) developed during this reporting period and on the viability assessment designs for the repository and engineered barrier.

2.3 PLANNED WORK TOWARD A LICENSE APPLICATION

Following completion of the viability assessment, the DOE will concentrate on work required to complete and submit the license application. The major actions needed to reach this objective include preparation of a final environmental impact statement prepared to requirements in the DOE's 10 CFR 1021; determination of site suitability according to the criteria in 10 CFR 960, and submittal of a site recommendation to the President as required by the NWPA; and development of a license application requesting construction authorization per the NRC's requirements set forth in the disposal regulations (currently 10 CFR 60).

The supporting site data, design data and analyses, and performance assessments to be used for each objective will represent the most current information available.

2.3.1 Environmental Impact Statement

The draft and final environmental impact statements will be developed using data and analyses from Project performance, design, and site investigation activities. A final environmental impact statement will be required if the Yucca Mountain site is recommended for approval to the President by the Secretary.

During FY 1998, site data will continue to provide input for the environmental description to be included in the environmental impact statement. Performance assessment activities will include thermohydrologic and engineered barrier system/waste form abstractions, biosphere abstractions, and system performance analyses. The preliminary draft environmental impact statement will be completed by the end of FY 1998, to be followed by an internal DOE review period. Before the end of FY 1999, DOE expects to distribute the draft environmental impact statement for public comment. The public comment period, which will include public meetings, is expected to be concluded in the first half of FY 2000. Public comments will be taken into consideration in preparation of the final environmental impact statement. The final environmental impact statement is planned for issuance by the Secretary of Energy near the end of FY 2000.

2.3.2 Site Recommendation

The Secretary of Energy may recommend the site to the President if the site is found suitable. This site recommendation would initiate a series of events that potentially lead to submittal of the license application to the NRC. These events involve consideration of the site recommendation by the President, Congress, and the State of Nevada. As the initiator of this process, the site recommendation is a key milestone on the path to licensing the proposed repository.

Before the Secretary recommends the site to the President, the DOE will determine the suitability of the Yucca Mountain site for development of a repository on the basis of the siting guidelines in 10 CFR 960. In December 1996, the DOE published a notice of proposed rulemaking to add a subpart specific to Yucca Mountain (see Section 1.4.1). The DOE believes the suitability of the site should be focused on how a repository at Yucca Mountain will affect the public health and safety. The proposal would focus the suitability decision on whether a repository built at Yucca Mountain would perform in a manner that would sufficiently protect the public.

To accompany a recommendation, the Secretary is to make available to the public, and submit to the President, a comprehensive statement of the basis of the recommendation, as described in NWPA Section 114. The site recommendation statement must include both information compiled by the DOE, and the views and comments of specific external organizations. The DOE information includes:

- Description of the proposed repository, including preliminary engineering specifications

- Description of the waste form or packaging proposed for use at the repository, and an explanation of the relationship between the waste form or packaging and the geologic medium of the site

- Discussion of data, obtained in site characterization activities, relating to the safety of the site

- Final environmental impact statement

- Any other information that the Secretary considers appropriate, such as transcripts from the consideration hearings.

External inputs to the site recommendation statement include:

- Preliminary comments of the NRC concerning the extent to which at-depth site characterization analysis and waste form proposal seem to be sufficient for inclusion in the license application

- Views and comments of the Governor and legislature of any state, or the governing body of any affected Indian tribe

- A report submitted by the State of Nevada on the economic, social, public health and safety, and environmental impacts likely to result from site characterization activities.

The Secretary must also respond to the views and comments of the states and affected tribes.

While the implementing organizations of the Project focus on the production of the viability assessment and its component parts, the planning organization has shifted to the development of the detailed strategy and implementation plan for the site recommendation in 2001. These plans and their component parts will be discussed in future progress reports.

2.4 LICENSE APPLICATION

The license application is the vehicle by which the Project will demonstrate to the NRC that the repository can and will be safely constructed, operated, and eventually closed, and that its postclosure performance will be acceptable. The license application will initially be submitted to the NRC in 2002 to seek authorization to construct the repository. Then, after issuance of the construction authorization and at the appropriate time during repository construction, an updated license application will be submitted to the NRC to seek a license to receive and possess radioactive materials at the repository. The Project schedule makes the assumption that this license will be issued by the NRC in 2010.

The *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain* (Management Plan) Revision 0 (YMP 1997e) was completed at the end of the reporting period and will be issued in October 1997. The Management Plan provides direction for the management framework and process to develop a license application to construct a repository at Yucca Mountain in compliance with the content requirements specified in 10 CFR 60.21.

The *Technical Guidance Document for License Application Preparation* (Technical Guidance Document) (YMP in prep. [a]) was issued in draft form in September 1997. When complete, the Technical Guidance Document (YMP in prep. [a]) will provide specific technical guidance to license application authors. Each chapter of the Technical Guidance Document (YMP in prep. [a]) corresponds to a chapter in the license application. The Technical Guidance Document will list applicable regulations that must be addressed; provide guidance and acceptance criteria based on the NRC's *License Application Review Plan* (NRC 1995) and regulatory precedent; and include content guidance from applicable Regulatory Guides, NUREG documents, and industry codes and standards. Analyses performed as part of the Compliance Program will provide information to the Technical Guidance Document (YMP in prep. [a]). Revision 0 of the Technical Guidance Document (YMP in prep. [a]) will be issued in FY 1998, and an additional revision will follow in subsequent years to ensure that the license application for construction authorization addresses all regulatory requirements and appropriate precedents at the time of submittal.

The Compliance Program is a formal program that is under development to assess regulatory guidance documents and industry experience to determine applicability to the design of the MGDS. This determination, along with supporting rationale, will be documented in Compliance Program guidance packages, which will in turn be used as input to the system description documents for the MGDS design basis. The guidance packages will also provide input to engineering design guides and to design analyses related to design basis events. The results of these analyses will be incorporated into the *Technical Guidance Document for License Application Preparation* (YMP in prep. [a]), and will be reflected in the license application. Conformance of the design with guidance provided in the Compliance Program guidance packages will ensure that the design is consistent with established regulatory precedent and will facilitate the NRC's timely review of the license application. The Compliance Program will also address use of regulatory guidance documents and industry experience in non-engineering topics important to licensing, such as radiological protection, emergency planning, and others.

The *Management Plan for the Development of a Project Integrated Safety Assessment (PISA)*, (YMP 1997f) Revision 2, was issued in July 1997. The DOE had planned to develop a Project Integrated Safety Assessment that would present a preliminary safety analysis report of site characterization results, including site investigation, repository and waste package designs, and performance assessment to support the viability assessment.

At the close of the reporting period, the DOE determined that there was no need to complete the safety assessment in its entirety. Instead, the chapters on the site description and total system performance assessment will be developed and completed as stand-alone documents to support the viability assessment. The DOE will develop the working draft license application in FY 1999 according to guidance in the *Management Plan for the Development of the License Application for a High-Level Waste Repository at Yucca Mountain* (YMP 1997e) and the *Technical Guidance Document for License Application Preparation* (YMP in prep. [a]). After completion of the working draft, license application development will continue to support development of the version that will be submitted to the NRC in 2002 to seek construction authorization.

The products completed for the viability assessment and site recommendation will contribute to development of the license application. Other planned activities that provide significant support for successful development of the license application are discussed in the subsections that follow.

2.4.1 Total System Performance Assessment for License Application (TSPA-LA)

The TSPA-VA will serve as the basis for development of the TSPA-LA. The TSPA-LA, which will be completed in 2000, will be based on continued refinement of the process models that support the performance assessment completed to support the viability assessment (TSPA-VA).

As deemed appropriate by the DOE, the TSPA-LA will reflect:

- Performance assessments and/or sensitivity analyses performed in support of DOE's site recommendation process, including development of its repository environmental impact statement

- External comments and observations addressing the TSPA-VA, including those generated by the TSPA Peer Review Panel, the Nuclear Waste Technical Review Board, and the NRC

- Additional site data that potentially impacts assumptions or results contained in process models abstracted into the TSPA-VA.

2.4.2 Issue Resolution

The NRC has stated that resolution of certain Key Technical Issues is of major importance to successful licensing of the proposed repository. Issue resolution with the NRC during this reporting period included several formal and informal interactions and continued development of a technical report on disposal criticality and seismic topical reports II and III. The Project is also referencing these Key Technical Issues as part of the viability assessment and will address them further as the Project prepares for submittal of the license application. The criticality technical report is discussed in Section 4.3.6.

In June 1994 the DOE submitted a topical report, *Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain* (Seismic Topical Report I, Revision 0) (YMP 1997a) to the NRC. Following NRC requests for additional information, review, and comment resolution, DOE issued Revision 1 of the topical report. The topical report, which incorporates all comments by the NRC staff, was completed and issued in August 1997.

Seismic Topical Report II, *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain* (YMP 1997b), was first submitted to the NRC in October 1995 as Revision 0. Revision 1 was submitted to the NRC for review in October 1996. DOE issued revision 2 of this topical report to the NRC in August 1997; this revision reflected NRC comments and concerns provided at a February 27, 1997, Appendix 7 interaction and March 21, 1997, correspondence. The DOE has not received NRC staff confirmation that all comments on the topical report have been resolved; however, the confirmation is expected in FY 1998.

An annotated outline for a third and final seismic topical report, *Determination of Preclosure Seismic Design Basis for a Geologic Repository at Yucca Mountain*, is in preparation, with the review draft scheduled for completion by December 31, 1997. Seismic Topical Report III will present the results of the application of the seismic hazard assessment methodology detailed in the first topical report and the seismic design methodology documented in the second topical report (YMP 1997b). An Appendix 7 meeting to discuss the annotated outline with the NRC staff is being planned for early 1998. The topical report is scheduled for completion in September 1998.

2.4.3 Web-Based Information System

The Web-Based Information System was under development during this reporting period. This effort, if proven feasible, is expected to allow results of work performed for the Project to be made available to the public. The Web-Based Information System may also represent a means to provide the NRC regulator with electronic access to the license application and safety analysis report, as well as electronic access to scanned images of supporting references (or their citations) available in the Project records systems. Testing of the Web-Based Information System resulted in refinements to the system and identified a need for increased emphasis on reference availability and traceability within the Project. Further testing of the Web-Based Information System will occur in FY 1998.

2.4.4 Mined Geologic Disposal System Design

As described in previous progress reports, design of the MGDS will be developed incrementally. Those structures, systems, and components that are important to waste isolation or radiological safety and have little or no regulatory precedent will be designed to the greatest level of detail first (Bin 3). Those structures, systems, and components that are important to safety or waste isolation but with regulatory precedent (Bin 2) will be designed to greater detail before license application submittal than those with no safety or waste isolation significance (Bin 1). The goal of the design effort to support the license application is to provide sufficient level of design detail to support the license application safety case for preclosure radiological safety requirements and postclosure performance requirements in 10 CFR 60. This design effort also includes supporting the TSPA-LA. During FY 1998, design work will directly support the viability assessment and license application, whereas the design work in FY 1999 will concentrate on license application design.

Emphasis has been placed on development of design bases early in the design process to assure efficiency in the development of design for the license application. This work has progressed in the *Mined Geologic Disposal System (MGDS) Requirements Document* (DOE 1997a) the *Controlled Design Assumptions Document* (CRWMS M&O 1997d), the Compliance Program, and the System Description Documents. This work is briefly discussed below. For more information on the design basis documents, see Section 4.1.1.

- A draft Revision 3 of the MGDS Requirements Document (DOE 1997a), which contains the system level requirements that the MGDS must meet to fulfill its mission to safely receive and dispose of high level waste, was prepared during this reporting period. Formal issuance of this draft as Revision 3 will occur in FY 1998.

- The Controlled Design Assumptions Document (CRWMS M&O 1997d) was updated twice during this reporting period. The document contains requirements, data, key project assumptions, and regulatory interpretations that must be verified but are captured to ensure this information is used consistently throughout the project.

- System description documents are under development that will contain design, construction, descriptive, and operational information for each of the systems that make up the MGDS. This reporting period's emphasis has been on development of Section 1 of the system description documents, *Functions and Criteria*. Of the 56 systems identified, drafts for Section 1 for 28 systems were completed during this period. Priority of development is assigned based on a previously developed binning system, described in Section 4.1.1. In FY 1998, initial versions of all the Bin 3 system description documents and a part of the Bin 2 items will be issued.

CHAPTER 3 – REPOSITORY PERFORMANCE

Repository performance assessment is the process of quantitatively evaluating the ability of the repository system and its components to contain and isolate radioactive waste according to applicable regulatory criteria. Performance assessment is used to evaluate preclosure radiological safety of the public and the workers and to evaluate the postclosure waste isolation capability of the repository system.

All three elements of repository performance assessment (preclosure radiological safety assessment, postclosure performance assessment, and performance confirmation) require the development of mathematical and analytical models of processes and events; whether induced by nature or by the construction and operation of the engineered components. The complexity of the interactions, as well as the necessary incompleteness of data, requires the incorporation of uncertainty and variability into each of the models. The following is a discussion of the work accomplished during this reporting period to assess the behavior of the repository system both in the preclosure and postclosure phases, as well as the plans to determine the adequacy of those analyses during the preclosure period.

Key Advances

Analysis of preclosure radiological safety – To conform to dose limits as described in 10 CFR 20, researchers analyzed design basis events in two categories of frequency. Preliminary results indicate that the repository meets the 10 CFR 20 requirements, with scenarios of potential bounds for a Category 1 spent fuel assembly drop or Category 2 rockfall or runaway transporter.

Development of the Total System Performance Assessment-Viability Assessment (TSPA-VA) preliminary base case – For postclosure performance assessment, analyses began that will develop the preliminary base case, which includes the current reference design discussed in Chapter 4 and to be presented in Volume 2 of the Viability Assessment product. Interpretation of the analyses of abstraction and testing workshops will contribute to the construction of tools, models, and parameters. A Total System Performance Assessment Peer Review Panel issued critical observations and recommendations to improve the total system performance assessment model. Design options being investigated include ceramic coatings on waste packages, ceramic drip shields, and backfill.

Analyses of TSPA-VA components – Because components are closely related, analyses began on natural and engineered components of the total system. Analyses will help develop coherent, predictive models. Of the models that are currently in development, the unsaturated-zone model addresses the concerns of uncertainties, alternative models, and reasonable ranges of parameters. Analyses are continuing for these models: unsaturated-zone thermohydrology, near-field geochemical environment, waste package degradation, waste form degradation, engineered barrier system radionuclide transport, unsaturated-zone transport, saturated-zone flow and transport, biosphere, and disruptive features, events, and processes. Processes and domains incorporated in the models listed here will contribute to the construction of the total system model.

Definition of activities to confirm and test that the planned repository meets performance objectives – The published Performance Confirmation Plan specifies how the Project will evaluate the repository's performance after closure. The plan supports performance described in the license application, and how it will be monitored, tested, and analyzed. Tests and evaluations have been defined and published, to conform to requirements in preclosure and postclosure periods.

3.1 PRECLOSURE RADIOLOGICAL SAFETY

This section discusses work accomplished in support of assessing preclosure radiological safety.

3.1.1 Objectives of Safety Assurance

The primary safety assurance focus was on development of the design basis events analyses for the repository preclosure operational period. The objective was to identify potential bounds for design basis events and to evaluate their radiological consequences. These analyses are in process and will be completed during the next reporting period.

Credibility and dose limits of 10 CFR 60 are frequency dependent, and design basis events are categorized by frequency (see Table 3-1).

Table 3-1. Frequency Category Definitions

Frequency Category	Category Definition
1	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.
2	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.
NC	These events are deemed to be not credible. As a result, no quantitative dose limits are applied.

The process involves determination of the frequency of an event to determine if the event is credible. If not, no quantitative dose limits are specified by 10 CFR 60, no further analysis is required, and there is no impact to other repository design or licensing organizations. If the event is determined to be credible, it is categorized based on the 10 CFR 60 definition, and an analysis is performed to determine if the dose limits associated with the applicable category can be met. Category 1 events (i.e., frequency greater than 1×10^{-2} events per year) require dose limits as specified in 10 CFR 20 for the public and occupational workers. Category 2 events (i.e., frequency less than 1×10^{-2} and greater than 1×10^{-6} events per year) require dose limits as specified in 10 CFR 60 for the public beyond the preclosure controlled area. The events being analyzed, current status, preliminary conclusions, and impact of the conclusions are presented in Table 3.2.

Table 3-2. Design Basis Event Analysis Summary

Event Analyzed	Status	Preliminary Conclusion	Impact of Conclusions
Aircraft Crash	Analysis in progress to estimate frequency of aircraft crash into surface facilities. The analysis is scheduled for completion in fiscal year 1998.	Event may be within Category 2 frequency range. Ongoing analysis will determine actual frequency and category of event	If categorized as a Design Basis Event, surface design may have to provide mitigation features
Industrial/Military Activity	Analysis in progress to evaluate the potential for objects to be dropped from military aircraft, and ongoing activities at Nevada Test Site and transportation-related and commercial activities (i.e., mining) that occur in the region. The analysis is scheduled for completion in FY 1998.	Event may be within Category 2 frequency range. Ongoing analysis will determine actual frequency and category of event.	If categorized as a Design Basis Event, surface design may have to provide mitigation features.
Rainstorm-Related Events	An analysis of rainstorms, debris avalanching, flooding, and landslides is being performed. Analysis will determine if an event scenario that could result in a radioactive release is credible or if any such release would be within the applicable dose limits. The analysis is scheduled for completion in FY 1998.	Underground facilities and portal entrances are above the probable maximum flood level. Radiologically controlled areas of the surface facilities may be located in the flood zone. Debris avalanching and landslides will not impact underground or surface facilities.	Input to surface design to require siting of radiologically-controlled buildings above probable maximum flood levels; to elevate floor areas above the flood levels; or provide diversion walls to deflect flood waters.
Rockfall/Ground Support Fall	An analysis is in progress to determine releases and consequences from an assumed rockfall within an open emplacement drift. The analysis assumes a rockfall or emplacement drift concrete liner strikes one waste package and causes a radionuclide release from one waste form. This analysis will be issued in December 1997.	Dose requirements are met at a 5-km boundary. Margin for meeting dose requirements increases at greater distances from the location of the radionuclide release due to dispersion. Using modeling assumptions, offsite dose requirements of 10 CFR 60 can be met following the design basis event.	Input from this consequence analysis will determine if high-efficiency particulate air filtration is required to enable subsurface operation to meet applicable radionuclide release limits in the event of a Category 2 design basis event.
Safeguards and Security Events as Design Basis Events	An analysis of the safeguards and security events and their relationship to design basis events is in progress. The analysis is scheduled for completion in FY 1998.	Events in this group will be addressed by the safeguards and security program and should not be included as credible design basis events.	Input will impact facility design, whether determined to be design basis events or part of the safeguards and security program.

Table 3-2. Design Basis Event Analysis Summary (continued)

Event Analyzed	Status	Preliminary Conclusion	Impact of Conclusions
Ashfall (from volcanoes)	An analysis is in progress to estimate the frequency and consequences of ashfall onto the surface facilities of the repository. The analysis is scheduled for completion in FY 1998.	The frequency of occurrence within a distance that could have an ashfall impact to the repository ranges from 5 X 10^{-8} to 3 X 10^{-10} events per year. Ashfall is not considered a credible design basis event.	No radiological consequence analysis is required, and there is no impact to other licensing or design organizations.
Spent Nuclear Fuel Canister Events (i.e., drop, slapdown, drop onto sharp object, and collision)	The worst-case event that is currently being analyzed is the drop of one vertically-oriented spent nuclear fuel canister from the fuel-handling crane onto another spent nuclear fuel canister located in an open shipping cask or disposal container. This scenario is analyzed for commercial reactor spent fuel assemblies. The analysis is scheduled for completion in FY 1999.	The net drop height, frequency, and the waste form have been determined by the ongoing analysis. The calculated frequencies are less than 1 X 10^{-2}per year, which classifies this event as a Category 2 design basis event in accordance with 10 CFR 60. Both spent nuclear fuel canisters are assumed to be damaged equally (i.e., both suffer damage assumed equal to falling through the drop distance onto a hard surface).	Input to surface design for a high-efficiency particulate air filtration system to mitigate dose to meet offsite dose limits at a preclosure controlled boundary.
Spent Nuclear Fuel Assembly Events (i.e., drop, slapdown, drop onto sharp object, and collision)	The worst-case event currently being analyzed is a drop of one vertically-oriented spent fuel assembly from the fuel handling crane onto another identical spent fuel assembly located in an open shipping cask, dryer, lag storage rack, or partially-filled disposal container. The analysis is scheduled for completion in FY 1999.	The calculated frequencies are greater than 1 X 10^{-2} per year, classifying this event as a Category 1 design basis event in accordance with 10 CFR 60. The unmitigated dose does not meet dose limits at the analyzed receptor locations (i.e., repository worker or public).	Input to surface design for a high-efficiency particulate air filtration system to mitigate dose to meet repository worker and offsite dose limits at a preclosure controlled area boundary.
High-Level Waste Canister Events (i.e., drop, slapdown, drop onto sharp object, and collision)	The worst-case event currently being analyzed is a drop of one vertically oriented high-level waste canister onto another canister. Both canisters are damaged equally (i.e., both suffer damage assumed equal to falling through the drop distance onto a hard surface). The analysis is scheduled for completion in FY 1999.	The calculated frequencies are greater than 1X 10^{-2} per year, classifying this event as a Category 1 design basis event in accordance with 10 CFR 60. The unmitigated dose does not meet dose limits for this design basis event at the analyzed receptor locations (i.e., repository worker or public)	Input to surface design for a high-efficiency particulate air filtration system to mitigate dose to meet repository worker and offsite dose limits at a preclosure controlled area boundary.

Table 3-2. Design Basis Event Analysis Summary (continued)

Event Analyzed	Status	Preliminary Conclusion	Impact of Conclusions
Transporter Runaway	The worse-case runaway is assumed to begin while traveling down the North Ramp into the repository, impact, and release particulate-form radionuclides. The analysis is scheduled for completion in December 1997.	The calculated event frequency is less than 1 X 10^{-2} and is classified as a Category 2 design basis event in accordance with 10 CFR 60. Using modeling assumptions, the unmitigated dose can meet limits at the analyzed offsite receptor locations.	Design input for the subsurface ventilation system to meet offsite receptor dose limits at a preclosure controlled area boundary.

3.1.2 Conclusions from Design Basis Event Analyses

Preliminary results from ongoing design basis event analyses indicate that a potential scenario with bounds for the Category 1 design basis events is a spent fuel assembly drop. With credit for a high-efficiency particulate air filtration system, the projected mitigated doses meet repository worker dose limits and public dose limits at a 5-kilometer preclosure controlled area boundary.

Assuming that the aircraft crash and industrial-military activity are not determined to be credible events, the potential bounds for Category 2 design basis events are runaway transporter and rockfall scenarios. Using numerous modeling assumptions, analyses show that Category 2 regulatory dose limits can be met at a 5-kilometer preclosure controlled area boundary without: (1) shutdown of the ventilation system, (2) high-efficiency particulate air filtration, (3) an elevated stack, or (4) rapid isolation of the tunnels to contain the releases. The validity and the applicability of these results will continue to be a focus in the development of the preclosure radiological safety basis.

The design basis event analyses in progress have used actuarial data from the literature and simplistic radionuclide release models to estimate scenario frequencies and radiological doses. Uncertainties exist in both design basis event frequency and radiological dose estimates that have not yet been quantified in the ongoing analyses. Therefore, these design basis event results are preliminary. Further analysis using evolving design features, operational constraints, and radionuclide release mechanisms will refine and may affect the final results. Uncertainty assessments will identify the compliance margins.

3.1.3 Classification of Mined Geologic Disposal Facility Structures, Systems, and Components

A classification analysis (CRWMS M&O 1997cc) has been completed for the MGDS. This analysis used descriptions in the current Mined Geologic Disposal Facility Architecture and the Systems Description Documents summaries to determine quality-affecting items based on their safety and waste isolation related functions. It also established the appropriate quality assurance classification(s) for those items. This analysis serves as the basis for revision of the Yucca Mountain Project *Q-List* (YMP 1997d), the continually evolving document that identifies the structures, systems, and components in the MGDS that have been determined to be quality affecting. This proposed revision to the *Q-List* aligns it with the current Mined Geologic Disposal Facility Architecture and provides classification requirements and guidance for ongoing design activities to ensure that radiological safety and waste isolation requirements are met.

3.2 POSTCLOSURE PERFORMANCE ASSESSMENT

Postclosure performance analyses are used to evaluate the effects of spatially and temporally varying processes that can alter the safety provided by the repository system after the system is sealed and closed. This section reports the development of the parameters and models that represent the various components of the entire system in these analyses. The studies that develop the basis for the parameters and models used to describe these components are reported in Chapters 4 (Design) and 5 (Site Characterization). This section also reports progress on the development of the total-system model architecture. Treatment of uncertainty in models and parameters is an important aspect of probabilistic total system performance assessment calculations. Specific sources of uncertainty are addressed in each of the sections below that discuss the various components of the total system performance assessment. Development of the methods and tools to address the uncertainty rigorously for the TSPA-VA will be performed in the next reporting period.

The primary goal of this activity is to construct the tools, models, and parameters required to perform the TSPA-VA. Results of the TSPA-VA will be used to identify the information needs still remaining to feed the Total System Performance Assessment-License Application (TSPA-LA). Thus, the interpretation of the analyses will provide information allowing the Project to direct and prioritize the site characterization and design activities most effectively.

The schedule for developing the TSPA-VA required completion of the abstraction/testing workshops by June 1997. The analyses started during this period will continue into the next reporting period. Results of the analyses that will provide the basis for the TSPA-VA preliminary base case will be delivered in the next reporting period. The preliminary base case will be run and results will be reported to the DOE on January 30, 1998. Preliminary documentation of the subsystem models used for the preliminary base case will be provided to appropriate Management and Operating Contractor organizations and DOE for review by March 1998. Sensitivity studies will be initiated immediately after the preliminary base case analyses are completed. A status presentation on the progress of the sensitivity studies will be provided to DOE by April 15, 1998. The final draft document of the TSPA-VA will be completed by June, 1998.

3.2.1 Objectives of TSPA-VA

In this reporting period, the primary performance assessment focus was on development of the TSPA-VA. The objective of the TSPA-VA is to incorporate the most current site and design models into a defensible representation of the probable long-term behavior of a potential repository system at Yucca Mountain.

The TSPA-VA is also expected to be a "dry run" for the analyses used to support the license application. TSPA-VA will be rigorously reviewed by internal and external review groups and by the NRC. Feedback from these reviews will be incorporated into the development and implementation of the TSPA-LA. In addition, the TSPA-VA will guide information needs from site characterization and design activities to adequately support the development of models for the TSPA-LA.

3.2.1.1 Plan and Schedule for the TSPA-VA

Planning of the TSPA-VA began in late FY 1996 and was reported in the *Total System Performance Assessment-Viability Assessment Plan* (CRWMS M&O 1996c). The general plan included steps for developing and enhancing the abstracted models for the various components of a total system performance assessment and also for developing and implementing the overall methodology and approach for the total-system modeling. However, information feeding total system performance assessment models is rapidly evolving as new site data are collected and interpreted. Changes to the potential repository system also occur as design alternatives are explored, incorporated, and/or discarded. Because of the highly synergistic nature of the engineered and natural components of the system, changes to any element are likely to change the response of other elements. Therefore, any plan developed for the TSPA-VA must also include the flexibility to assess, reprioritize, and alter modeling activities to ensure the best possible product.

During this reporting period, the majority of the effort associated with the TSPA-VA activity has been directed at developing abstracted models for use in the total system performance assessment to describe the different natural and engineered components of the total system. Sections 3.2.2 through 3.2.12 report the results of these activities. Another major effort is associated with combining these components into a total system representation that will allow projections of probable behavior. The total system representation will include alternative conceptual models, variability and uncertainty in parameters and processes, and a sufficient level of detail to maintain the sensitive aspects of each process. Section 3.2.13 reports progress for this activity.

Activities associated with construction of the TSPA-VA will be divided into two major elements: (1) development associated with a preliminary base case (which will be completed on January 30, 1998) and (2) the subsequent sensitivity analyses and revision of the base case for the final TSPA-VA analyses. The preliminary base case will be constructed using the reference design, as defined on September 30, 1997, (discussed in Chapter 4) and a stochastic representation of the expected features, events, and processes. Thus, reasonably narrow distributions will be developed for the preliminary base case sampling. Disturbances, such as nuclear criticality, volcanism, and seismicity, will not be included. After completion of the preliminary base case, additional sensitivity analyses will be performed to explore the effects of widening certain distribution ranges, incorporating additional conceptual models, including alternative design features (such as backfill and drip shields), and including disturbances to the system. The information obtained through these sensitivity studies will be incorporated into the TSPA-VA analyses to best represent the Project's interpretation of the range of probable behaviors. Figures 3-1 and 3-2 show the generalized schedule for completion of abstraction/testing and TSPA-VA.

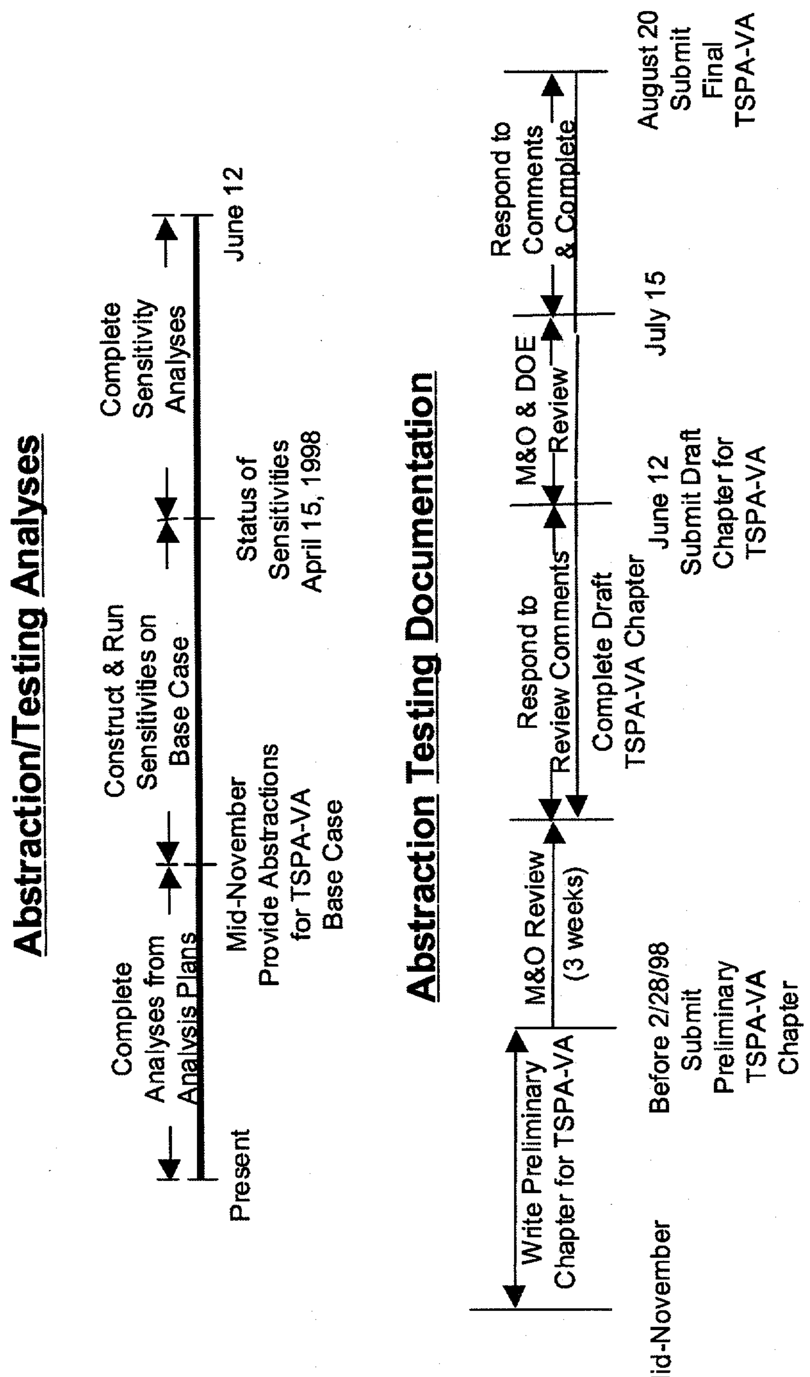

Figure 3-1. Schedule for Abstraction/Testing Activities Analyses and Documentation

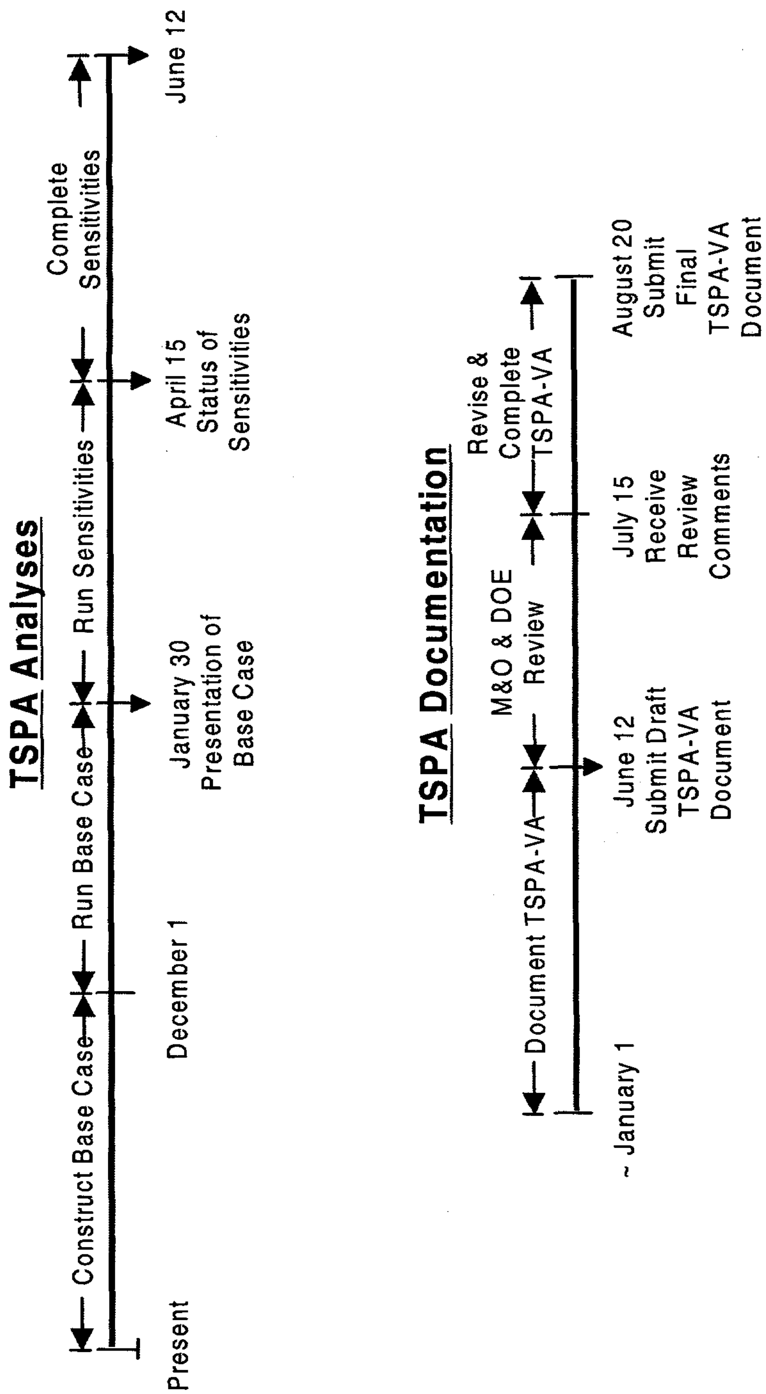

Figure 3-2. Schedule for TSPA Analyses and Documentation

A detailed discussion of the plans for the TSPA-VA was contained in *Total-System Performance Assessment-Viability Assessment Methods and Assumptions* (CRWMS M&O 1997cd).

The Total System Performance Assessment Peer Review Panel, which was formed in the previous reporting cycle, is expected to provide continual feedback on ways to improve the total system performance assessment model results and documentation during TSPA-VA development and implementation. Their first interim report was provided to the Project on June 20, 1997. A meeting to discuss their comments further was held in Las Vegas on July 9, 1997. The next interim report will be completed in the next reporting period. The general observations and beliefs of the Panel were as follows:

- The Project lacked a current overall description or conceptual model of how the repository is expected to perform during future time periods
- The Project should place more emphasis on understanding fracture flow
- The Project was not adequately addressing the mechanical and chemical aspects of coupled effects in the Drift Scale Test
- No working model or set of models was seen to exist for the prediction of the performance of the engineered barrier system and the waste package
- There appeared to be a paucity of Project-specific experimental data for the engineered barrier system and the waste package
- The repository must be analyzable to be licensed
- The Project treatment of the transport of radionuclides from the waste package appeared to be ultra-conservative (which would lead to high estimated release rate)

The presence of thermally hot conditions might unnecessarily increase the complexity of the repository performance assessment analysis

The results of the waste form degradation model must be testable, traceable, and transparent
The NRC needs to clarify the applicability of its current criticality regulation to postclosure performance

The Project should meet with the EPA to obtain any guidance that may be available until such time as the standards are issued

The Project could benefit from timely input and informal discussions with outside experts.

3.2.1.2 Repository and Engineered Barrier System Design for TSPA-VA

In general, the major components of the base case, or current reference, design will include an areal mass load of 85 MTU/acre, with point loading of the waste packages (see Section 4.2.1.3).

The engineered barrier system will include a drift lining (concrete), an initial air gap in the drift (no backfill), a two-layer waste package (10-cm thick corrosion allowance material and 2-cm thick corrosion resistant material), in-drift emplacement of the waste packages, placement of the waste packages on a steel pedestal, and a concrete invert at the base of the drift. Information on the current repository reference design is in Section 4.2.

3.2.1.3 Design Option Sensitivity Cases

The viability assessment design is expected to include the preliminary base-case design as well as several design options. The highest priority options currently being investigated (CRWMS M&O 1997v) are ceramic coatings on the waste packages, ceramic drip shields supported by the waste packages, and backfilled emplacement drifts. Backfill alone is not regarded as one of the highest-priority options. In combination with ceramic coatings or drip shields, it is primarily intended to protect the ceramics from rockfall. Another potential design option is cladding credit, i.e., taking credit for potential protection of spent fuel by its cladding (zircalloy, in most cases). Therefore, inclusion of cladding credit in the calculations will be one type of sensitivity analysis performed for TSPA-VA. Figures 3-3 and 3-4 indicate the locations of these design options.

The following design sensitivity cases will be evaluated:

- The reference design (which includes no ceramic coatings or drip shields, and no backfill) without cladding credit (the TSPA-VA preliminary base case)

- The reference design with some degree of cladding credit, perhaps sampled probabilistically

- A design option in which waste packages are coated with ceramic (to provide additional protection against corrosion) and emplacement drifts are backfilled. (A backfill material has not yet been chosen.) Tuff gravel and quartz sand are currently under consideration. Perhaps both will be analyzed, splitting this case into two cases. Cladding credit is included in this case

- A design option in which waste packages are protected by ceramic drip shields (low-permeability ceramic designed to deflect water seeping into emplacement drifts away from waste packages) that are supported by the waste packages, and emplacement drifts are backfilled. (See comment about backfill material.) Cladding credit is included

- A design option that includes both ceramic waste-package coatings and waste package-supported drip shields, plus backfill. Cladding credit is included.

Additional design options may be analyzed if time permits. Some options of interest are the use of alternative metals for the corrosion-resistant materials, backfilled drifts without use of ceramic coatings or drip shields, line loading rather than point loading (that is, emplacing the waste

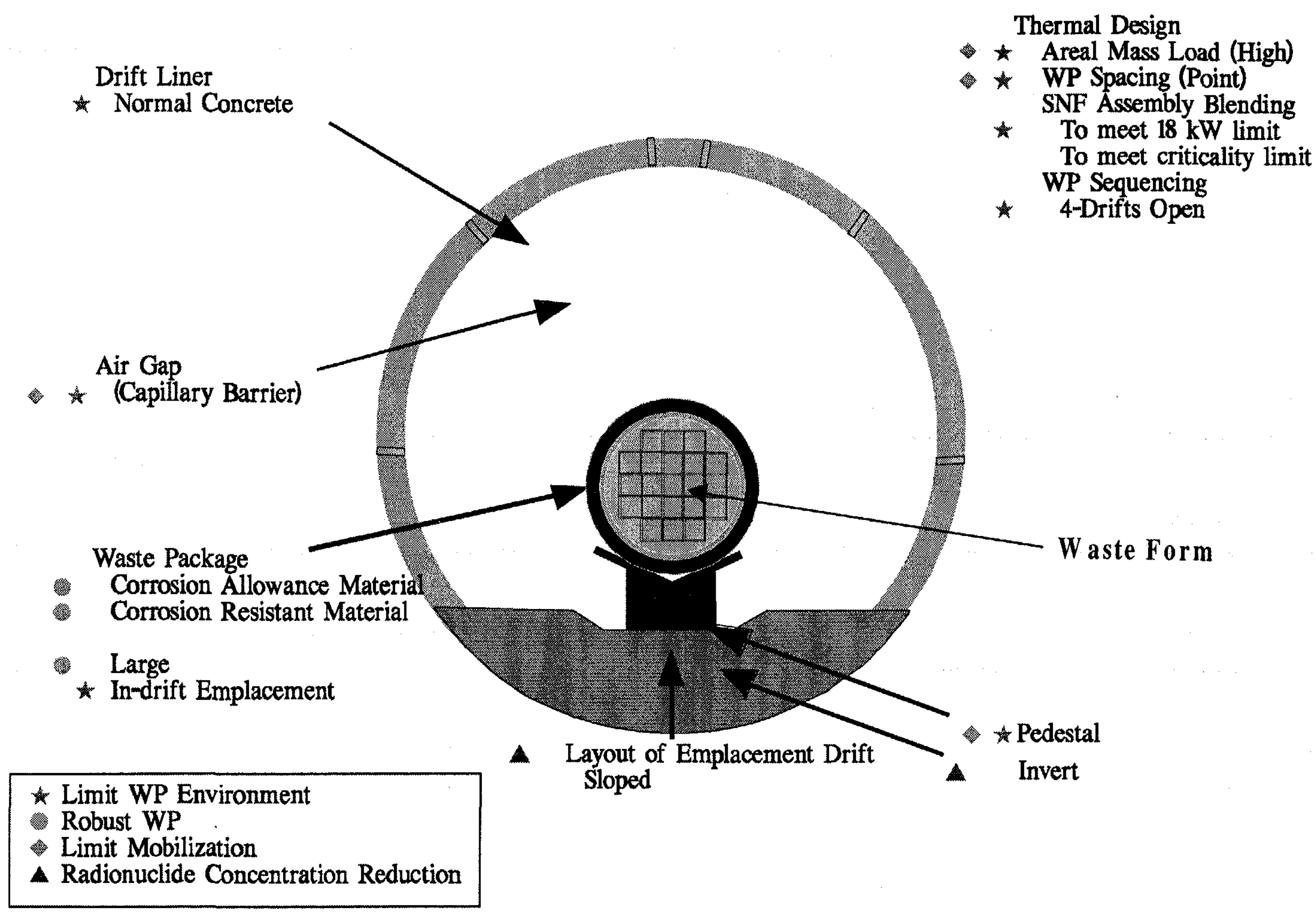

Figure 3-3. Base Case Design for TSPA-VA (after M&O 1997bb)

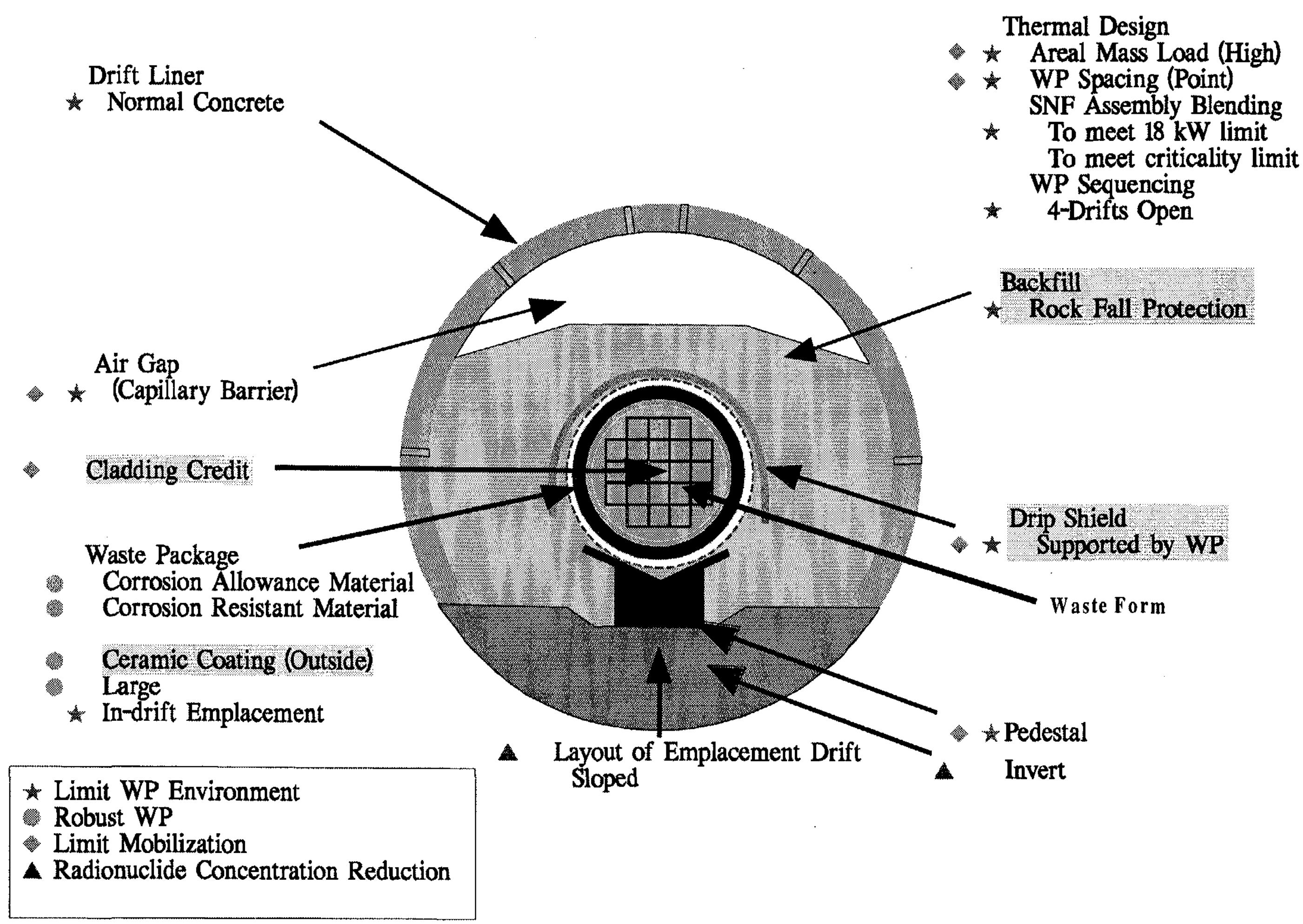

Figure 3-4. Design options presented on Base Case Design (after M&O 1997b)

packages very close together in order to equalize temperature and humidity between hotter and cooler packages), and the use of additives in backfill or invert in order to manipulate the geochemical environment.

3.2.2 Basic TSPA-VA Components

The TSPA-VA is expected to consider all relevant features, events, and processes that could significantly influence the ability of the system to contain and isolate radionuclides from the biosphere. Each component included in such an analysis will be chosen based on either its relatively high likelihood of occurrence or its potentially relatively high consequence.

Total system performance assessment-type analyses are done for the repository system because there are essentially no independent components. All the components are closely interrelated. Changes to one aspect almost invariably have feedback effects on one or more of the other components. However, the extreme complexity of the system has led the Project to divide the system along certain boundaries at which the features, events, and processes change in some fundamental and significant way.

The basic components used for development of the TSPA-VA are unsaturated-zone flow, unsaturated-zone thermohydrology (subdivided into mountain-scale and drift-scale), near-field geochemical environment, waste package degradation, waste form degradation and radionuclide mobilization, engineered barrier system transport, unsaturated-zone transport, saturated-zone flow and transport, and biosphere. In addition, disturbances to the natural system resulting from basaltic volcanism, seismicity, and nuclear criticality are also considered.

Development of the performance assessment models and associated parameters and assumptions during this reporting period is discussed for each of these basic components in Sections 3.2.3 through 3.2.13.

3.2.3 Abstraction of Unsaturated-Zone Flow Model

During this reporting period, documentation (CRWMS M&O 1997cm) was completed on the unsaturated-zone flow model workshop conducted in the last reporting period, and analyses based on this documentation were begun. In addition, an expert elicitation conducted to evaluate the Project data and models for unsaturated-zone flow was reported during this period.

The unsaturated-zone flow model for the Project (Bodvarsson et al. 1997) is being developed. The goal for the model is to synthesize all the available data into a coherent, predictive model of water and air flow in the unsaturated zone. The available data are too sparse to identify a unique calibrated model, and a number of calibrated models can fit the data equally well. One of the functions of performance assessment is to consider the uncertainty introduced by such multiple models and to show how they translate into uncertainties about repository performance. Therefore, much of the emphasis in the unsaturated-zone flow modeling for TSPA-VA is on uncertainties, alternative models, and reasonable ranges of parameters.

The current unsaturated-zone flow model uses the dual-permeability formulation of fracture and matrix flow and interaction. This is a continuum formulation, with a matrix continuum, a fracture continuum, and an interaction between them that is proportional to the pressure difference at each spatial location. The numerical formulation of the model uses the TOUGH2 computer code. Model calibration is done using an inverse method implemented in ITOUGH2 to optimize the model parameters (matrix and fracture hydrologic properties for 22 hydrogeologic units). In the model used as the preliminary base-case unsaturated-zone flow model for TSPA-VA, flow is mostly through fractures in the welded layers and mostly through the matrix in the non-zeolitized nonwelded layers. A new feature of the latest model is that a fracture-matrix coupling parameter, related to the wetted area of contact between fractures and matrix in each hydrogeologic unit, is used as an inversion parameter as well. It is important to note that spatially variable infiltration at the surface in the current model is considerably higher than values that were being used just two years ago. The higher infiltrations come from the latest infiltration model (Flint et al. in prep.), which is based primarily on neutron-hole data. Infiltration is not allowed to vary in the ITOUGH2 inversions, but infiltration uncertainty is explored to some extent by performing multiple inversions, with the infiltration map multiplied by different factors. The increase in infiltration estimates has important implications for repository performance.

The Project recently conducted an unsaturated-zone flow model expert elicitation, in which Project data and models were presented to a group of experts for their evaluation of the work being done (CRWMS M&O 1997ce). Particular emphasis was placed on estimates of surface infiltration and deep percolation; probability density functions of infiltration and percolation were elicited from each expert. The individual probability density functions plus a mean probability density function are presented in the 1997 report (CRWMS M&O 1997ce). The information derived from the elicitations will be used to supplement the models that have been developed within the Project.

The process modelers and site characterization data gatherers have strongly recommended that the Project use three-dimensional models to represent Yucca Mountain flow adequately. This recommendation is based primarily on flow beneath the potential repository, where there is thought to be significant nonvertical flow because of heterogeneity in the locations of the zeolitic layers and perched water. From parameter-sensitivity results reported by Bodvarsson et al. (1997) and from additional sensitivity studies that have been done, the key unsaturated-zone flow parameters are the fracture matrix connection area reduction factor, the fracture van Genuchten alpha parameter, and infiltration. Also potentially important are matrix and fracture permeabilities and the fracture van Genuchten μ parameter (also, for transport and possibly for thermohydrology, the fracture porosity), but these parameters appear to be less important than the first three listed above. Variations on the most important parameters will be evaluated in order to determine their impact on repository performance.

An alternative unsaturated-zone flow model will be defined that is expected to be similar in behavior to the Weeps model, which was used in previous Yucca Mountain system analyses (Barnard et al. 1992; Wilson et al. 1994). Measured and inferred values for all matrix and fracture hydraulic properties will be used, and the fracture-matrix reduction factor (fmx)(the degree to which the modeled interaction between the fracture and matrix is decreased, thus allowing fracture flow) will be proportional to upstream relative permeability. (Discussion of this choice for fmx is presented in Bodvarsson et al. (1997)). This alternative model could provide useful insights into repository performance. It will show whether repository performance is adversely affected by a much greater fast-flow component than exists in the preliminary base-case model and will provide information on performance for thermal and climate changes.

In addition to large-scale unsaturated-zone flow, a model of seepage of water into emplacement drifts is needed. Drift-scale modeling of water flow and seepage is being performed to investigate the conditions under which water will seep into a drift, and the amount of seepage that might be expected. Those modeling results, as well as past results from the Weeps model (CRWMS M&O 1996a, Chapter 6), and some conceptual ideas for estimating weep spacings from the mountain-scale dual-permeability formulation, will be used to define a response surface abstraction for seepage. Although there is a high degree of uncertainty in the models, additional field data collection and drift-scale modeling is underway (reported in Section 5.1).

The preliminary base case unsaturated-zone flow for TSPA-VA will be represented using the Project unsaturated-zone flow process model with some parameter variations. The parameter variations will include three to five spatially variable infiltration maps and a limited variation of key hydrologic parameters within ranges that would not adversely affect the model calibration. The mean probability density function for infiltration that was developed as part of the unsaturated-zone flow model expert elicitation (CRWMS M&O 1997ce) will be used to weight the results for different infiltration values.

The preliminary base case will also include climate changes, present conditions (dry climate, mean infiltration approximately 5 mm/yr), the long-term average climate (infiltration on the order of three to four times the present rate), and super pluvial conditions (infiltration possibly as much as 20 times the present rate). The infiltration increases and the timing of the climate changes will be based primarily on paleoclimate information.

3.2.4 Abstraction of Unsaturated-Zone Thermohydrologic Model

During this reporting period, documentation was completed on the unsaturated-zone thermohydrology model abstraction/testing workshop (CRWMS M&O 1997ci) conducted in the previous reporting period. Analyses based on this documentation were begun. Initial planning was done to implement an expert elicitation on coupled processes (including thermal, chemical, and mechanical processes) in the rock immediately adjacent to the drift. The elicitations are anticipated to begin in the next reporting cycle.

Current understanding of unsaturated-zone thermohydrology is not as advanced as the understanding of ambient unsaturated-zone flow. Thermally driven effects have only recently been included in total-system analyses, typically using a simplified format. However, the Project's understanding of unsaturated-zone thermohydrology continues to grow on an experimental level with the advent of the ESF thermal testing program. The thermal testing program includes the small-scale single-heater test and the large-scale drift-scale test. A large block heater test has also aided understanding and supported the design of the underground heater tests. Among other things, the thermal testing program offers an opportunity for process-level conceptual-model validation and provides information on the coupled thermohydrologic processes occurring as a result of heat input to the geologic setting. Unsaturated-zone thermohydrologic models at the scale of the mountain and drift have grown in complexity with increasingly more attention paid to physical details. Higher dimension numerical models are now possible for idealized conceptual flow models. Model dimensionality is an important consideration in the thermohydrologic simulations of the potential repository.

Current drift-scale models (both thermohydrologic and conduction-only) are typically specified as two- or three-dimensional. Two-dimensional models are computationally very efficient, but they neglect important heat-transfer resistances associated with the finite areal size of a waste package (CRWMS M&O 1996c). Three-dimensional models explicitly account for the areal extent (i.e., the true surface area) of the waste package and include the resistance to heat transfer associated with the surface area of the heat-generating source. Three-dimensional models are recommended at this scale. The obvious drawback of the three-dimensional drift-scale model is the reduction in computational efficiency. All the drift-scale models mentioned above use the matrix-satiated equivalent continuum model concept for heat and fluid flow in fractured porous media. TSPA-VA drift-scale thermohydrologic models will make use of an alternative formulation of the equivalent continuum model with a reduced matrix satiation value to ensure fracture flow of the liquid phase during the condensate shedding periods. Condensate shedding in fractures was observed in the single-heater test and is inferred from the large block test.

An important assumption in the thermohydrologic modeling is that mechanical and chemical alterations of the hydrologic properties (i.e., coupled thermohydrologic-mechanical and thermohydrologic-chemical effects) can be neglected. This assumption is being tested by sensitivity studies. If these effects are found to be significant, they will be included in the TSPA-VA simulations in a simplified manner (presumably by modifications of matrix and fracture permeability), but detailed inclusion would have to be left to future work (i.e., TSPA-LA). Information on these effects should be available from the single-heater test in time for use in the TSPA-VA.

Drift-scale thermohydrologic calculations will predict temperature and relative humidity at the surface of the waste package and liquid saturation and temperature in the invert and the drift lining. These quantities will be provided for representative repository locations and for different waste package types. The NUFT computer program (Nitao 1996) is being used for drift-scale thermohydrologic calculations. The gas-phase flow rate and air mass fraction in the drifts will be calculated using both drift-scale and mountain-scale thermohydrologic models in order to determine whether mountain-scale gas flow affects them. The TOUGH2 computer program (Pruess 1991) is being used for mountain-scale thermohydrologic calculations.

The Project is planning a thermohydrology expert elicitation for FY 1998, which may provide some additional guidance on treatment of issues related to the coupling of thermal, mechanical, hydrologic, and chemical effects in the host rock immediately adjacent to the drift wall.

The TSPA-VA preliminary base case for unsaturated-zone thermohydrologic calculations will include information from subsurface repository design, waste package design, and site characterization. Subsurface design provides detailed design data related to the emplacement drift geometry, total thermal load, and mode of waste package emplacement. Waste package design provides additional information related to waste package placement strategy and fuel types as well as the waste stream information. Finally, site characterization provides the data related to geologic stratigraphic, hydrologic- and thermal-property data, infiltration rates applied at the ground surface, and climate change.

The preliminary base-case stratigraphic and hydrologic properties have been derived from the Project unsaturated-zone flow process model (Bodvarsson et al. 1997). In order to maintain consistency among all TSPA-VA models, the preliminary base-case properties and parameters developed in the unsaturated-zone flow task are applied to the thermohydrologic models as well. Specifically, these parameters include detailed hydrologic property data sets for fractures, matrix, and fracture-matrix interaction. Thermal properties have been obtained from a compilation of Project experimental results and models. Unsaturated-zone thermohydrology will be incorporated into TSPA-VA simulations by means of a series of tables of near-field flow rates and thermodynamic state variables. Although initial modeling assumes constant boundary conditions, the preliminary base case for TSPA-VA includes climate change, which will be included in the thermohydrologic models by means of step changes in the boundary conditions at specified times.

The TSPA-VA preliminary base case assumes that the emplacement drifts will remain intact for a period of hundreds to perhaps a thousand years, after which the drifts will be filled with rockfall rubble (Bhattacharyya in prep.). The change in the thermal properties of the drifts actually caused by this rubble will taken into account in the drift-scale thermohydrologic models, unless the change can be shown to have little effect.

Sensitivity cases for unsaturated-zone thermohydrology are expected to include the alternative repository designs. Variations in fracture-matrix coupling are not meaningful if the modified equivalent continuum model is used for thermohydrologic calculations and the unsaturated-zone flow sensitivity cases. Alternative repository designs of interest include backfill and line loading. Work now in progress, described in Section 5.1, will provide an increasingly more comprehensive basis for the thermohydrologic modeling that supports the total system performance assessment.

3.2.5 Abstraction of Near-Field Geochemical Environment Model

During this reporting period, documentation was completed on the near-field geochemical environment workshop (CRWMS M&O 1997cj) conducted in the previous reporting period, and analyses were begun based on this documentation.

The near-field geochemical environment model represents the environment inside the drift formed as a result of the interactions among various elements of the natural and engineered systems. The near-field geochemical environment model captures the combined effects of the chemistry of the in-drift materials, the thermohydrologic response of the repository, and the amount and composition of gas and aqueous phase materials entering the drift. The form and evolution of the near-field geochemical environment model is critical because of its influence on waste package degradation and waste form alteration and mobilization. Components formed in the near field may also be transported out of the drift and have an effect on the far-field flow and transport characteristics.

The near-field geochemical environment model in past total system performance assessments has primarily focused on the impact of pH and temperature variations. However, a full description of the near-field geochemical environment, including evolution of the aqueous, gaseous, solid, and colloidal phases and the microbial communities, has not been accomplished. The near-field geochemical environment model developed for the TSPA-VA will be a major improvement over past representations. However, the limited amount of information and models available for the near-field geochemical environment will restrict the sophistication of the final model to support TSPA-VA. Work is in progress, as described in Section 5.1, to provide additional scientific support to this modeling for the period after the viability assessment.

The near-field geochemical environment model is composed of four abstraction models: (1) a water-solids chemistry model; (2) a colloid-facilitated radionuclide transport model; (3) a model addressing the influence of microbial environments; and (4) a model for gas composition evolution. The primary model affecting the total system performance assessment representation of the engineered barrier system is the water-solids model, which supplies the fluid compositions through time that react with the ground support materials, waste packages, the waste forms, and the underlying engineered barrier system components, and the composition of water entering the geosphere for transport. The colloid model will be used to assess releases of plutonium transported via colloidal iron-oxyhydroxides. Results from the models of microbial communities and the gas composition evolution will be fed mainly back through the water-solids chemistry model. Results generated by the water-solids chemistry model for use in the total system performance assessment model will be abstracted engineered barrier system solids evolution scenarios and fluid composition response surfaces for a number of regions of the engineered barrier system.

The time dependence for the near-field geochemical environment model will be a coarse representation consisting of pre-boiling, boiling, and post-boiling. For the preliminary base case analyses, the fluid and gas compositions entering the drift are being constrained consistent with the thermohydrologic scenarios that provide in-drift temperature history, air mass fraction of in-drift gases, and gas and water fluxes into the drifts. Therefore, the preliminary base-case gas composition will reflect the measured ambient pore gas compositions (approximately atmospheric except for carbon dioxide which is set to about 1000 ppm in the pore gases), and the thermal effects during the boiling period. Work on modeling the water compositions sampled from the single-heater test provides some constraints on the evolution of the gas composition in the heated rock system. The preliminary base-case assumption is that this gas composition is imposed throughout the drift. The composition of the fluid coming down fractures and entering the drift will be represented by a 90 percent condensate and 10 percent ambient fluid equilibrated with this gas composition.

Within the drift, concrete and steel are materials that affect the fluid composition as it moves through the drift. For the preliminary base case, no alteration of the gas phase due to these interactions is incorporated. Both the concrete and steel will have scenarios of alteration mineralogy defined for the three time periods given above. Fluids will be equilibrated with these mineral scenarios and the gas composition for each time period, with the resultant fluid becoming the starting composition for the next location in the drift. The preliminary base case locations are just outside the drift and also within the concrete lining, the corrosion products, the waste form (provided by the waste form degradation model), the corrosion products, and the concrete lining. The resulting final fluid composition is then provided to the unsaturated-zone transport modeling group as the composition of water leaving the drift.

For the TSPA-VA preliminary base case analyses, a colloidal plutonium source term and transport model will be included. This will be the first direct incorporation of colloids into the Project total system performance assessment analyses. The model addresses clay and iron-oxyhydroxide colloids because they are assumed to have the largest potential to affect radionuclide release from the system. Literature data will be used to constrain the stability of iron-oxyhydroxide colloids and experimental data will be used for constraints on the sorption processes. The third model will provide constraints on the effects of microbial communities on the near-field geochemical environment model by calculating the magnitude of microbial activity through time as limited either by energy or nutrients. The products of this plan are a spatial-temporal description of pH; the temporal generation of carbon dioxide; and the temporal evolution of microbial population (i.e., biomass) as affected by nutrient availability, relative humidity, temperature, microbial reaction rates, and initial microbial community. The model for gas evolution addresses the gas compositional changes driven by the thermohydrology in four gas constituents (water vapor, carbon dioxide, oxygen, and nitrogen) for the preliminary base case.

The primary variations on the preliminary base case for near-field geochemical effects are expected to include modifications to incorporate specific features of the unsaturated-zone system at Yucca Mountain. Examples are the modification of the in-drift gas composition due to the reaction with introduced materials.

3.2.6 Abstraction of Waste Package Degradation Model

During this reporting period, documentation was completed on the waste package degradation workshop (CRWMS M&O 1997ck) conducted in the previous reporting period, and analyses based on this documentation were begun. Additionally an expert elicitation on waste package degradation was begun during this period reporting.

The primary role of the waste package degradation model is to provide information on the lifetime and overall degradation of the waste package. The degradation description is to be developed as a function of the environment of the waste package. This model is very significant in the overall total system performance assessment, because it contributes to the determination of the rate and time of the release of radionuclides from the waste package.

The current subsystem model for evaluating the degradation of the waste package is the WAste Package DEGradation (WAPDEG) model (CRWMS M&O 1997cf). The WAPDEG model currently evaluates the degradation of the waste package outer barrier using an empirical formulation developed from experimental data. The evaluation of nickel-based inner barrier degradation is based on the model developed from an expert elicitation. Individual corrosion models for the carbon steel outer barrier and the nickel-based inner barrier continue to be developed and are, in the future, to be based more on the mechanistics of corrosion processes. First-cut models of microbial induced corrosion for the barriers are being developed. These models, once completed or updated, will be incorporated into the WAPDEG model.

For modeling outer barrier corrosion, both humid air and aqueous corrosion (as a function of exposure time, temperature, and relative humidity), will be simulated. Localized variations of outer barrier corrosion are represented by a pitting factor as a multiplier to the uniform general corrosion depth. The model also includes the spalling of corrosion products from the outer barrier (Lee et al. 1997). The primary inputs to the model are temperature and relative humidity (at the waste package surface) as a function of time; waste package design (such as the corrosion allowance material and corrosion-resistant material barrier thickness); and pitting factor for the localized corrosion of the outer barrier. The updated outer barrier model is intended to include dependencies on the water chemistry, salt-scale formation, and dripping water.

The current model assumes that the inner barrier is only subject to aqueous pitting corrosion and calculates the distribution of pitting rates as a function of temperature. Also, a simple galvanic protection model has been used, which only allows pitting corrosion of the inner barrier after a certain percentage of the outer barrier thickness has been corroded. This galvanic protection model will be significantly revised for the TSPA-VA and will not be included in the preliminary base case. The updated WAPDEG model is expected to include pitting corrosion, active crevice corrosion, and the interaction between the inner and outer barriers. These interactions include crevice formation between the outer barrier and the corresponding outer barrier corrosion products and the inner barrier; pH suppression in the crevice due to the hydrolysis of dissolved metal ions from corrosion of both the barriers, galvanic coupling between the barriers, and accumulation of corrosion products inside the crevice.

Microbial-induced corrosion was not explicitly included in previous total system performance assessment waste package degradation analyses, though it could be argued that it was embedded in the corrosion data from natural exposure conditions, which were used to develop corrosion models for the carbon steel outer barrier in WAPDEG. In TSPA-VA, the microbial-induced corrosion will be modeled as localized corrosion incorporating additional constraints due to temperature, water availability, nutrient availability, and pH.

The waste package degradation expert elicitation provided distributions for the following parameters:

- Temperature threshold for the corrosion of the corrosion allowance material outer barrier in the presence and absence of salts

- Relative humidity threshold for humid air corrosion in the presence and absence of salts, dust, oxides, and corrosion product spalling

- Relative humidity threshold for aqueous corrosion in the presence and absence of salts and oxides

- Pit density, diameter, and aspect ratios, assuming high pH conditions

- General corrosion and pitting rates for the corrosion-resistant material inner barrier
- Pitting rates and corrosion localization factors for the corrosion allowance material.

The panelists from the waste package degradation expert elicitation also provided qualitative input on corrosion morphology, behavior, and geometry. They commented on the effect of drips, corrosion modes, modeling approaches, microbial induced corrosion, radiolysis, and other issues useful in the formulation of conceptual models and sensitivity studies. The distributions provided by the waste package degradation elicitation results will provide a partial basis for development of the TSPA-VA model and will be used in conjunction with other information from within and outside the Project, as appropriate.

The primary issues of waste package degradation include appropriate representation of the corrosion of the corrosion allowance material and the corrosion-resistant material, humid air and aqueous effects on corrosion, and microbiological effects on corrosion rates. Additionally, uncertainty in several aspects of the system must be evaluated, including juvenile failure of the waste packages, degradation of the drift configuration leading to rockfall on the waste packages, and the long-term structural integrity of the waste packages. The spatial variability of the environment in the drifts will also contribute to variable degradation states within the waste package population. Sections 4.3.5.1, 4.3.5.2, and 5.1 describe ongoing work to address these uncertainty sources.

The preliminary base-case WAPDEG output will be corrosion or degradation time-histories of the waste packages for a given design and multiple repository environments, addressing the two major performance goals of the waste packages, i.e., waste containment and waste isolation (DOE 1996b). The waste containment goal will be measured as the time until waste package failure which is defined as first pit or crack perforation.

Sensitivity analyses will be conducted on numerous features of the waste package degradation model to evaluate their importance to waste package performance. In particular, many of the abstractions provided by the waste package degradation expert elicitation will be evaluated in a sensitivity framework. Some of the key sensitivity analyses to be conducted include:

- Patches (size of the area under corrosive attack)
- Critical relative humidity switches for corrosion
- Pitting factor of carbon steel
- Dripping rates
- Chemistry effects
- Galvanic protection
- Microbially induced corrosion
- Stress corrosion cracking of the inner barrier
- Ceramic coating on waste package
- Ceramic drip shield
- Backfill

- Mechanical failure of degraded waste package by rockfall
- Welds

3.2.7 Abstraction of Waste Form Degradation Model

During this reporting period, documentation was completed for the waste form degradation workshop (CRWMS M&O in prep. [a]) conducted during the previous reporting period; analyses based on this documentation were initiated. Initial planning began for an expert elicitation on waste form processes to begin implementation in the next review period.

Three general categories of waste forms will be incorporated into the TSPA-VA: commercial spent nuclear fuel, defense high-level vitrified waste, and DOE-owned spent nuclear fuel (including Navy spent nuclear fuel). Alternative degradation models are being developed to evaluate the waste form degradation. The commercial spent nuclear fuel degradation model for the TSPA-VA is a response surface from the empirical model being developed by O'Connell (Interdepartmental Memo). The model, based on spent fuel dissolution testing results, has functional dependencies on fuel burnup, pH, and temperature. The defense high-level waste glass dissolution model for the TSPA-VA is a response surface from the empirical model developed by Stout and Leider (1997). The model has functional dependencies on temperature and pH. The DOE spent nuclear fuel will be modeled as a composite waste form and will require development of an appropriate degradation model. The models reported in Duguid et al. (1997), will be reviewed to ascertain the most appropriate model for the evaluation.

Another aspect of waste form degradation is cladding degradation. A cladding model is being developed that addresses initially failed cladding, cladding failure due to mechanical conditions, cladding failure due to temperature effects (uniform surface oxidation), and cladding failure due to creep. The cladding model will provide a surface area exposed as a function of time.

The major issues associated with the waste form degradation identified in the waste form abstraction/testing workshop (CRWMS M&O in prep.[a]) include:

- Evolution of the near field environment and its effect on waste form degradation
- Development of appropriate cladding and waste form degradation models
- Importance of secondary mineral precipitation effects
- Significance of vapor hydration of defense high level waste
- Representation of data uncertainty and variability
- Mechanical failure modes for cladding.

It is expected that the available data from waste form degradation testing will be processed and incorporated into a response surface with important functional dependencies. These models will be developed for commercial spent nuclear fuel, defense high-level waste, and DOE spent nuclear fuel.

The response surface from the commercial spent nuclear fuel model will be incorporated as a functional form into the total system performance assessment code. Depending on the sensitivity of performance to the fuel burnup term, the response surface may only include dependencies on pH and temperature. The response surface from the defense high-level waste model will be incorporated as a functional form into the total system performance assessment code. The functional form for this material will include dependencies on temperature and pH. The composite waste form for the DOE spent nuclear fuel will be incorporated as a single waste package group into the total system performance assessment code. Should it prove necessary to include more detail than an average or composite waste form can provide, a release term for each of the key DOE spent nuclear fuel waste forms will be developed external to the total system performance assessment code and input as a source term in the total system performance assessment code. The cladding model will include a component for juvenile failure, creep rupture, and mechanical failure of the cladding. This model will be implemented in the total system performance assessment code as waste form surface area exposed as a function of time.

The primary sensitivity cases to be evaluated for waste form degradation and mobilization include:

- Varying the dissolution rates for alternative dissolution models to account for uncertainty in rates caused by lack of data on geochemical effects and secondary phase formation

- Evaluating the performance variance caused by different fuel burnup values to determine the need for inclusion into the model

- Varying the surface area exposed due to cladding failure to account for the uncertainty in many aspects of the cladding model that can cause failure.

3.2.8 Abstraction of Engineered Barrier System Radionuclide Transport Model

Documentation was completed this period for the engineered barrier system radionuclide transport abstraction/testing workshop (CRWMS M&O in prep. [a]) which was conducted during the previous period. Analyses based on the documented results from the workshop were begun. Ongoing work on the technical basis for flow and transport modeling in the unsaturated zone is described in Section 5.2.

The engineered barrier system as addressed in performance assessment is comprised of all the components of the repository within the drift. The overall transport of radionuclides within the engineered barrier system is influenced by waste package degradation, waste form degradation (including cladding degradation), the thermohydrologic environment, the chemical environment, and the design of the engineered barrier system. The radionuclide release from the engineered barrier system is then transported to the natural system for ultimate transport through the unsaturated zone and on to the saturated zone.

Radionuclide concentrations mobilized from the waste form are constrained by the solubility limit of each radionuclide. The solubility limits were formatted into distributions by expert elicitation reported in previous total system performance assessments (Barnard et al. 1992; Wilson et al. 1994; CRWMS M&O 1995). In some cases, the solubility constraints were described as a function of temperature or pH. Dissolution rates of the waste forms (spent nuclear fuel and vitrified defense high-level waste) were derived from experimental data using the flow-through technique, and were used in the previous total-system evaluation (CRWMS M&O 1995).

The solubility constraints for several radionuclides have been reevaluated in 1997. In particular, the neptunium solubility constraint has been updated based on additional information still in development (see section 5.4). Along the transport pathways in the engineered barrier system, the radionuclide concentrations are checked against the solubility limit.

The major issues associated with engineered barrier system transport, reported in Waste Form Degradation and Mobilization Abstraction/Testing Workshop Results (CRWMS M&O in prep. [a]), are:

- Important water contact modes
- Important transport paths and the sorption properties along those paths
- Colloidal fraction expected to be released and transported from the waste forms
- Influence of temperature, solid, and fluid dependencies on the solubility limits for the radionuclides released from the waste forms
- Appropriate degradation scenario for the engineered barrier system and its representation in the engineered barrier system transport evaluation

Determination of the parameters and scenarios, as well as the uncertainty associated with the issues, will provide a significant part of the information required for the abstraction of the engineered barrier system transport model. The configuration of the engineered barrier system and how it is expected to change with time will be the framework for the engineered barrier system transport model. Many of the model details were still in development during this reporting period.

The preliminary base-case abstraction of the engineered barrier system model in TSPA-VA will be developed using the total system performance assessment code compartment model. The engineered barrier system will be discretized to represent appropriately the waste form, waste package, corrosion products, and invert. In addition, upstream boundary conditions that reflect the degradation of the engineered barrier system (e.g., lining, rockfall) will be included. The process model to be used in TSPA-VA is intended to be input directly into the total system performance assessment code and to make use of the compartment or cell modeling capability within the total system performance assessment code.

Numerous engineered barrier system transport model sensitivity analyses are planned for TSPA-VA, including:

- Discretization of the compartments representing various components of the engineered barrier system

- Diffusion coefficients
- Liquid saturation in engineered barrier system components
- Water film thickness
- Colloidal transport
- Conceptual model uncertainties
- In-drift water seepage
- Waste package seepage
- Waste package failure mode
- Waste package premature failure
- Engineered barrier system material sorption properties

3.2.9 Abstraction of Unsaturated-Zone Transport Model

In this reporting period, documentation of the unsaturated-zone transport workshop was completed (CRWMS M&O 1997cg). Analyses using the documented workshop results have begun.

In past total system performance assessments (Barnard et al. 1992; Wilson et al. 1994; CRWMS M&O 1995), unsaturated-zone transport parameters found to have a major impact on overall system performance were (1) the unsaturated-zone percolation rate, (2) the partitioning of flow (and thereby radionuclides) between rock matrix and fractures, and (3) the radionuclide distribution coefficients between liquid and tuff matrix. The retardation of radionuclides is a very significant contributor to the natural system performance in that it significantly lengthens the travel time of almost all the radionuclides released from the engineered barrier system. However, a few radionuclides are not noticeably retarded through interaction with the host rock (such as ^{99}Tc, ^{129}I, and ^{36}Cl), and these become the most important contributors to peak dose at the accessible environment. Also, ^{237}Np is only slightly sorbing to the host rock, and because of its relatively lower solubility, it tends to be released from the engineered barrier system over much longer periods of time than ^{99}Tc and ^{129}I. Therefore, in past total system performance assessments, ^{237}Np has dominated long-time releases. This has made ^{237}Np the focus of much investigation within the unsaturated-zone radionuclide transport models. Another radionuclide of potential significance to unsaturated-zone radionuclide transport is ^{239}Pu, because of the possibility of transport in colloidal form. This method of transport has not been investigated in past total system performance assessments, but as discussed in this section, will be the focus of sensitivity studies in TSPA-VA.

The major issues associated with unsaturated-zone transport fall into three major categories: physical transport processes, chemical interactions and repository-perturbed environment, and heterogeneity (CRWMS M&O 1997cg). Among the important issues identified with treatment of the physical transport processes are the conceptual representation of the fracture-matrix interactions, the approach used to incorporate short- and long-term transients, identification of key fracture and matrix properties, and whether lateral dispersion is important. The second category includes issues such as the appropriate minimum distribution coefficient approach, importance of colloids, thermochemical alteration of minerals, the natural geochemistry, its evolution from the repository emplacement, adsorption in fractures, and influence of the thermohydrologic-mechanical changes. Finally, for heterogeneity, the issues include the details of the stratigraphy above and below the repository, distribution of zeolites, spatial distribution of infiltration, and the dimensionality of the models.

The unsaturated-zone radionuclide transport abstraction approach for TSPA-VA is to couple the FEHM (Finite Element Heat and Mass) (Robinson et al. 1997) code's particle tracker directly to the total system performance assessment code. The approach will model unsaturated-zone transport using a three-dimensional site-scale transport model (Robinson et al. 1997) and the three-dimensional unsaturated-zone flow fields generated by the site-scale flow model (Bodvarsson et al. 1997). In previous Project total system performance assessments, transport was represented using the total system performance assessment code's Markovian algorithm for interaction between matrix and fractures in a given hydrogeologic unit. This is a descriptive, rather than a process, model that must be verified against a process model such as finite element heat and mass. Also, all the unsaturated-zone flow and transport in the previous Project total system performance assessment (CRWMS M&O 1995) was represented with a set of six parallel one-dimensional vertical columns that covered the repository area and went from the repository horizon to the water table. (This type of geometric/dimensionality abstraction neglects lateral flow.) The proposed approach for TSPA-VA is fully three-dimensional and, therefore, does not have the problems with dimensionality abstraction encountered in previous Project total system performance assessments.

The preliminary base-case unsaturated-zone transport model uses a dual-permeability flow and transport formulation with advective and diffusive fracture-matrix interaction, linear sorption in the matrix, and radionuclide chain decay. Additionally, the preliminary base-case transport model will use the particle-tracking method for a two- and/or three-dimensional system and steady unsaturated flow. The preliminary base case will also include the parameter variations identified for the unsaturated-zone flow. with the remainder of the parameter ranges to be identified in the sensitivity calculations listed below. Additional sensitivity and model abstraction work will also be performed considering transport processes not included in the preliminary base case.

Sensitivity analyses for the unsaturated-zone flow and transport model will include:

- Fracture-matrix interaction
- Transient flow and transport
- Colloid-facilitated radionuclide transport

- Sorption models for radionuclide transport
- Effects of dispersion and fine-scale heterogeneity on radionuclide transport

3.2.10 Abstraction of Saturated-Zone Flow and Transport Model

A workshop on saturated-zone flow and transport was conducted in April 1997, and results of this workshop were documented (CRWMS M&O 1997e). Analyses based on the workshop documentation have begun. Also, the Project conducted an expert elicitation starting in May 1997. The results of this elicitation will be completed and documented in the next progress report. Ongoing work to further develop the scientific basis for this component is described in Section 5.3.

For the purpose of guiding TSPA-VA analyses, a prioritized list of technical issues related to groundwater flow and transport in the saturated zone was developed at the April 1997 Saturated-Zone Flow and Transport Abstraction/Testing Workshop (CRWMS M&O 1997e). These issues were prioritized on the basis of specific criteria linked to potential impact to repository performance. High priority issues in Category 1, Conceptual Models of Saturated-Zone Flow, included regional discharge and recharge, vertical flow, and alternative conceptual models. In Category 2, Conceptual Models of Saturated-Zone Geology, issues included channelization in vertical features and stratigraphic features, properties of faults, distribution of zeolites, and fracture network connectivity. In Category 3, Transport Processes and Parameters, issues included dispersivity, matrix diffusion, and sorption. In Category 4, Coupling to Other Components of total system performance assessment, issues included climate change, unsaturated-zone and saturated-zone coupling, thermal and chemical plumes, and well withdrawal scenarios.

In May 1997, an expert elicitation concerning saturated-zone flow and transport modeling was convened. Issues addressed and parameters estimated included:

- Conceptual model of saturated-zone flow
- Conceptual model of the large hydraulic gradient
- Specific discharge in the saturated zone below and downgradient of the repository
- Groundwater velocity in the saturated zone below and downgradient of the repository
- Hydraulic conductivity
- Anisotropy of hydraulic conductivity
- Change in water table elevation in response to climate change
- Dispersivity
- Effective porosity
- Sorption coefficients
- Eh and pH
- Contaminant dilution factor.

Final documentation of the elicitation is due in the next reporting period. The TSPA-VA analyses will incorporate the results of the expert elicitation to check the abstractions; to qualitatively compare estimates of uncertainties to the assumptions or numerical modeling results used in total system performance assessment; and possibly to provide input to the total system performance assessment Monte Carlo realization, distributions of hydraulic conductivity, dispersivity, effective porosity, and sorption coefficients.

For TSPA-VA, the saturated-zone preliminary base case will address features, events, and processes that have been identified in the Abstraction/Testing workshop and by the expert elicitation. Flow and transport in the saturated zone will be based on the detailed three-dimensional modeling by Czarnecki et al. (in prep.) and the three-dimensional transport model (Zyvoloski et al. 1997) as abstracted by the convolution integral method. Multiple simulations will be conducted to investigate uncertainties in parameters and in changes anticipated to result from future climates. Boundary conditions for the three-dimensional models, for both present and future climates, will be based on the regional saturated-zone model by D'Agnese et al. (in prep.). Some parameter variations that will be incorporated in the three-dimensional calculations, include different permeabilities, dispersivities (longitudinal and transverse), sorption coefficients.

Another form of abstraction will be used for estimating the dilution of radionuclide concentrations at groundwater withdrawal wells. This dilution factor will be based on three-dimensional pumping simulations; for the preliminary base case, the location of the water well will be at a distance of 20 km from the potential repository. The factor will be used to divide the simulated radionuclide concentration at points of interest, and it will be treated as an additional uncertain parameter.

For TSPA-VA, sensitivity analyses will be conducted to look at alternative conceptual models of the saturated zone. A listing of presently planned sensitivity analyses follows:

- Effect of channelization of flow
- Effects of repository heating and chemical changes
- Colloidal transport of plutonium
- Match between the methodology for total system performance assessment calculations and transient calculations
- Alternative conceptual models of flow
- Hydraulic properties of faults
- Vertical flow
- Unsaturated-zone flow and transport coupling with the saturated zone
- Well withdrawal scenarios.

3.2.11 Abstraction of Biosphere Model

In June, 1997, the Project conducted an abstraction/testing workshop on biosphere modeling. Results from the workshop have been documented (CRWMS M&O in prep.[f]). Analyses based on this documentation have begun.

The biosphere, as defined by the National Academy of Sciences, is "the region of the earth in which environmental pathways for the transfer of radionuclides to living organisms are located and by which radionuclides in air, ground water, and soil can reach humans to be inhaled, ingested, or absorbed through the skin. Humans can also be exposed to direct irradiation from radionuclides in the environment." The focus of the work reported in this section is to develop appropriate models to analyze the effects of releases of radionuclides from the potential repository at Yucca Mountain on a human population.

A workshop to determine the important issues for biosphere model development for the TSPA-VA, and courses of action to address these issues, was held in Las Vegas, Nevada, on June 2-3, 1997 (CRWMS M&O in prep. [f]. Participants produced a list of potential issues prioritized according to three criteria: (1) the effect on individual dose, (2) the effect on population dose, and (3) the range or uncertainty that the issue implied for determining biosphere dose conversion factors. Issues that were prioritized highest were to be addressed further if possible, either by abstraction into a model for TSPA-VA or by investigation as the subject of a sensitivity study. (Because of the short period of time between the workshop and date when biosphere results would be needed for inclusion in TSPA-VA, the fall of 1997, it was recognized that not all issues could be addressed.) Further details of this workshop and the resulting analysis plans are contained in the *Biosphere Abstraction/Testing Workshop Results* report (CRWMS M&O in prep. [f]).

Important issues for Category 1, Critical Group, included habits and location of the critical group and extrapolations of habits and location to the future. Important issues for Category 2, Climate, included effects of climate change on biosphere pathways, effects of climate change on critical group, and build-up of contaminants in soil. Important issues for Category 3, Pathway Variability, included location and definition of biosphere-geosphere interface, range of uncertainties and variability in parameters and pathways for critical group, variation in dominant pathways with time, radionuclides of importance, volcanism, radionuclide transfer in disruptive scenarios, and inadvertent intrusion.

Three analysis plans were developed to address as many important issues as possible and as practical in the time available before TSPA-VA. The analysis plans were:

- Provide appropriate biosphere dose conversion factors to total system performance assessment – this plan would provide the biosphere abstraction for the TSPA-VA

- Determine the radionuclides that are significant contributors to dose – based on a sensitivity study and previous total system performance assessments, a list of radionuclides would be developed for use in calculating biosphere dose conversion factors

- Provide consistency between radionuclide concentrations in the saturated zone and biosphere scenarios – the geosphere-biosphere interface would be defined in order to make adjustments to the concentrations of radionuclides in the geosphere at the point of contact with the biosphere; for example, well pumping would be investigated as a source of dilution.

For TSPA-VA, the biosphere will be abstracted by means of biosphere dose conversion factors. Biosphere dose conversion factors are the multipliers that convert a radionuclide concentration at the geosphere-biosphere interface into a dose to a human. The units are in terms of dose per concentration, such as rem/mg/^{1} or Sv/Bq/m^3. Biosphere dose conversion factors are radionuclide-dependent, scenario-dependent (where a scenario is a collection of pathways), and climate-dependent. Implicit in the use of biosphere dose conversion factors is that dose varies linearly with concentration at the geosphere-biosphere interface. Biosphere dose conversion factors will incorporate dose conversion factors, the multipliers that convert an amount of radiation to a dose (dose conversion factor units are in terms of dose per activity, e.g., rem/Ci, Sv/Bq, etc). For TSPA-VA, biosphere dose conversion factors will be calculated probabilistically and presented as a distribution reflecting some measure of their uncertainty. When used in a TSPA-VA calculation, a biosphere dose conversion factor will be sampled from this distribution for each realization.

The preliminary base case for TSPA-VA will consist of a reference adult person living 20 km from Yucca Mountain. The reference person will be defined using the survey of the existing population. Water for drinking and for food production will be from a well intercepting the highest concentration point (at 20 km from Yucca Mountain for the preliminary base case) of a plume of contaminated groundwater from Yucca Mountain. Concentration will be reduced at the well head because of mixing with uncontaminated water during pumping; the reduction will be determined in an ancillary calculation and will depend on the pumping rate. Radiation dose will be calculated (using biosphere dose conversion factors) for all significant radionuclides. The 12 radionuclides presently being considered are C-14, Cl-36, Se-79, Tc-99, I-129, Pb-210, Ac-227, Th-229, Pa-231, Np-237, Pu-239, and Pu-242. These radionuclides were determined to be the major contributors to dose using present total system performance assessment models and published tables of generic screening factors, and were estimated to account for 99.9 percent of the radiation dose at any time of concern.

Sensitivity analyses concerning the biosphere will be performed for TSPA-VA, if time allows. Features, processes, and events that might be investigated using the sensitivity studies include:

- The distance that the reference person lives from Yucca Mountain, e.g., 5 km and 30 km

- The dose from disruptive events, e.g., atmospheric contamination and surface contamination caused by volcanism and possibly by drilling

- The effect of climate change on dose, e.g., reduced irrigation, dust, and soil build-up

3.2.12 Abstraction of Disruptive Features, Events, and Processes Models

During this reporting period, documentation was completed for the disruptive features, events, and processes workshop that was conducted in the previous reporting period. Analyses based on this documentation (CRWMS M&O 1997ch) were begun.

Disruptive events are treated separately from the TSPA-VA preliminary base case. The response of the repository system to such disruptions is measured by the sensitivity of the results (e.g., doses) to the magnitude and type of disruption. Disruptive events to be considered in TSPA-VA include magmatic intrusions, seismic events, nuclear criticality, and human intrusion. Of these four classes of disruptions, most of the effort in FY 1997 was devoted to nuclear criticality. Previous total system performance assessment analyses (Barnard et al. 1992; Eslinger et al. 1993; Wilson et al. 1994) have investigated several volcanic and human-intrusion disturbances, while seismicity was treated as a follow-on calculation to the CRWMS M&O 1995 effort (Gauthier et al. 1996). Criticality has not been treated in prior total system performance assessment analyses. Human intrusion will not be treated explicitly in the TSPA-VA calculations, however, a sensitivity analysis is planned to investigate the consequences of a scenario where drilling would allow a direct release to the saturated zone.

Work was completed on nuclear criticality disturbed scenarios. Analyses and inputs from experts identified several scenarios that could result in the formation of potential critical configurations inside degraded waste packages. Additionally, scenarios for the formation of potential critical configurations in the near field (immediately outside the waste package) and in the far field were developed, but their likelihood is considerably less than that for in-package criticality. The criticality-scenarios analyses identified potential critical configurations but did not evaluate the impacts on repository system performance. That aspect of the work is scheduled to be completed for the TSPA-VA analyses in FY 1998.

A systematic description of the processes leading to in-package criticality is presented in the analysis. The design and engineering of the waste packages prevents criticality from occurring before the waste package is penetrated and water enters. Water in the waste package causes corrosion and degradation of the waste form and of the neutron-control structures incorporated into the waste package design; it also provides neutron moderation necessary for criticality for most of the fissile material expected to be emplaced. The various waste types and waste package designs have different modes of failure before they can form critical configurations. Further waste package failure and transport of fissile materials to the surrounding rock is required for the formation of near-field and far-field critical configurations. The primary obstacle to formation of critical configurations outside the waste package is the necessity to reconcentrate the fissile material at a site where a critical configuration could form. Most realistic geochemical analyses indicate that fissile material concentrations sufficient for criticality cannot be achieved by natural chemical processes. The impacts to repository performance of an extremely unlikely underground criticality will be evaluated in TSPA-VA.

The DOE has commissioned expert elicitations for both volcanism and seismicity. The Probabilistic Volcanic Hazards Analysis has been completed, while the Probabilistic Seismic Hazards Analysis is ongoing. The Probabilistic Volcanic Hazards Analysis has provided a probability distribution of the frequency of occurrence for magmatic intrusions inside the repository block. This information, combined with consequence analyses is used to develop the risk-based measure of repository performance. Similar information will be developed from the Probabilistic Seismic Hazards Analysis probability density functions for frequency of occurrence. Descriptions of continuing scientific work that will aid future assessments are in Section 5.8.

The following general classes of disruptive events are expected to be incorporated into final TSPA-VA (however, they are not part of the preliminary base case):

- Direct releases of waste to the surface from volcanic events, with an ash plume as the distribution mechanism – impact of such releases will be measured by inhalation and ingestion doses to a critical group

- Increase in the radionuclide source term for groundwater flow and transport resulting from the accelerated release of waste from waste packages damaged by a nearby magmatic intrusion

- Increase in the radionuclide source term for groundwater flow and transport resulting from the accelerated release of waste from waste packages damaged by seismic events, such as rockfall, faulting, or ground motion

- Changes in the groundwater flow and transport resulting from a seismic event occurring in the flow path outside the repository block

- Changes in doses to a critical group resulting from waste being mechanically transported in bulk to the saturated zone by a drilling incident

- Nuclear criticality incidents, as described above.

  Abstractions of all the disturbed scenarios that suggest increases in the groundwater-transport source term (i.e., the close-proximity-dike volcanic scenario, the seismic rockfall scenarios, and the in-package criticality scenarios) combine several disciplines. Primarily, the models reflect various forms of waste-package degradation resulting from the disturbances. The results of the sensitivity studies (the consequences) will be combined with probabilities of occurrence to provide risk measures consistent with the TSPA-VA preliminary base-case results. These can then be expressed as alterations to the final base-case results.

3.2.13 Description of Total System Model

The total system model will be constructed by synthesizing the various abstractions described in the previous sections. Numerous computer codes will be implemented in the TSPA-VA. Information or abstractions are fed to the integrating total system performance assessment code. The site-scale unsaturated-zone flow process model (implemented in the TOUGH2 code) (see Section 5.2) will be used to generate flow fields, and a particle-tracker in the FEHM code (see Section 5.2) will generate information for the unsaturated-zone transport calculations. TOUGH2 will also provide the parameter sets used by other process model codes to perform thermohydrologic calculations (see Section 5.1. The EQ3/6 and AREST-CT codes will be used to develop parameters for the near-field geochemical environment. The waste package degradation model, WAPDEG (see Section 3.2.6), will use input from the thermohydrologic and near-field geochemical environment evaluations.

The primary basis underlying the total system performance assessment code architecture is the domain-based or spatial abstraction. The repository system is naturally divided into a series of sequentially linked spatial domains, such as the waste package, the drift, the host rock, etc. This division is most useful for tracking the projected path, both in time and space, of the radionuclides from the waste form to the biosphere. The sequential order for information flow is site-scale thermohydrology → drift-scale thermohydrology → waste package degradation → engineered barrier system transport → unsaturated-zone flow and transport → saturated-zone flow and transport → biosphere transport. Thus, the input boundary conditions of each successive component in the sequential order is provided by the preceding one.

Both uncertainty and variability are intrinsic in the total system performance assessment models. The two tools that will be used to deal with uncertainty and variability are (1) alternative conceptual models and (2) probability theory. Uncertain processes will be represented with alternative conceptual models and uncertain parameters will be described as probability distributions. Monte Carlo analyses will be used as the primary method of analyzing uncertainty by calculating numerous realizations of the repository system. The range of calculated values for performance measures together with the probability for each one will be used to estimate total-system repository performance.

In addition to evaluating performance and associated uncertainty for the reference repository design and of various alternative designs, another important goal of the TSPA-VA will be to determine which components of the engineered and natural system have the most influence on performance. Therefore, TSPA-VA will also be used to provide information on which parameters, if better defined or better understood, would result in the greatest gain in confidence in the Project's understanding of repository performance (see steps 5 and 6 in the subsequent discussion of the Performance Confirmation approach, described in Section 3.3.3).

3.3 PERFORMANCE CONFIRMATION

The Performance Confirmation Program begins during site characterization and continues until permanent closure. In this program, data will be collected and analyzed to reduce uncertainties in postclosure performance and verify the performance of the structures, systems and components important to waste isolation. The Performance Confirmation Program is part of the Mined Geologic Disposal System Test and Evaluation Program that performs verification for the entire MGDS.

3.3.1 Performance Confirmation Plan

The *Performance Confirmation Plan* (CRWMS M&O 1997f) completed during this reporting period defines the activities necessary to conduct the Performance Confirmation Program as specified in 10 CFR 60, Subpart F. This plan specifies monitoring, testing, and analysis activities to be conducted for evaluating the accuracy and adequacy of the information used in the license application to determine that the performance objectives for the period after permanent repository closure will be met. This plan is to be used as a basis for detailed planning of the Performance Confirmation Program and is to be integrated with the MGDS Test and Evaluation Program. The Performance Confirmation Program objectives are to:

- Confirm that subsurface conditions encountered and changes in those conditions during construction and waste emplacement operations are within the limits assumed in the license application

- Confirm that natural and engineered systems and components required for repository operations, or that are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated

- Evaluate compliance with NRC postclosure performance requirements

- Evaluate the repository readiness for permanent closure.

The performance confirmation approach consists of six major steps:

1. Define a performance confirmation baseline that identifies the processes and parameters important to postclosure performance

2. Project values and variations of critical performance measures for the parameters in the performance confirmation baseline to establish expectations during construction and operations

3. Establish tolerances or limits on deviations from predicted performance

4. Monitor performance, perform tests, and collect data

5. Analyze and evaluate the data (using information to perform process model validation, analysis, statistical tests, and total system performance assessments)

6. Recommend and implement appropriate actions (if there are deviations from what was predicted or assumed), or when the closure criteria are satisfied, evaluate repository readiness for permanent closure.

The parameters and concepts identified for performance confirmation are based on the current understanding of natural and engineered barrier processes. This understanding is represented by numerical and analytical models developed to simulate these processes. However, uncertainties still exist in the parameters and models used to represent these processes. As new understanding is gained from continued site characterization, design, and performance assessment activities, the way in which these are simulated will change. This alteration will require changes in the list of performance confirmation parameters and in the *Performance Confirmation Plan* (CRWMS

M&O 1997f). A task is planned for FY 1998 to reassess the performance confirmation parameters based on the viability assessment design and sensitivities of parameters in the TSPA-VA. Recommendations from this assessment could reduce the scope of the planned program.

The *Performance Confirmation Plan* establishes requirements for testing and contains requirements on facilities and equipment necessary to perform the testing functions. An initial set of these requirements was captured in the *Controlled Design Assumptions Document* (CRWMS M&O 1997d). The Performance Confirmation Plan recommended some changes to these requirements. The design group has used these requirements from the Controlled Design Assumptions Document to develop designs of performance confirmation facilities and systems.

A strategy for mapping during repository construction was recommended in the Performance Confirmation Plan. The recommendation is to (1) map approximately 10 percent of emplacement drifts, based on the current drift spacing and layout; (2) map non-emplacement drift openings; and (3) observe rock mass conditions for anomalous conditions during construction. The rationale for mapping 10 percent of the emplacement drifts is that the frequency of mapped drifts is selected to assure intersection of features anticipated to affect repository performance. Present surface mapping shows several faults with 200 to 300 meter trace lengths within the repository block. Most of these faults are expected to penetrate the host repository horizon and extend downward to the water table. The importance of these faults to repository performance is currently uncertain. A frequency of mapping approximately 10 percent of the emplacement drifts, at the current spacing, would provide reasonable confidence of intersecting these surface mapped features at depth.

3.3.2 Mined Geologic Disposal System – Viability Assessment Test and Evaluation Plan

The *Mined Geologic Disposal System Test and Evaluation Plan* for the viability assessment (CRWMS M&O 1997g) was completed during this reporting period. This plan defines the processes, organizational structure, and test and evaluation functions necessary to support system design and development, verify compliance with requirements, evaluate the operational suitability and effectiveness of the repository system, and support the implementation of regulatory requirements. This plan discusses how the tests are defined, designed, conducted, and how data from the tests are evaluated against performance, functional, design, and regulatory requirements. A test organization is defined, establishing test support and working groups to oversee and manage the Test and Evaluation Program functions, which include:

- Determining the suitability of the Yucca Mountain Site for a geologic repository
- Conducting proof of design concept testing to reduce development risk
- Verifying compliance of structures, systems, and components with design requirements and specifications
- Performing system testing to validate system requirements including the receipt, handling, retrieval, and disposal of waste

- Conducting periodic performance testing to verify compliance with preclosure requirements

- Performing modeling, testing, and analyses to verify compliance with postclosure regulatory requirements.

Of the five MGDS Test and Evaluation Program major functional elements (Site Characterization Test and Evaluation, Developmental Test and Evaluation, Operational Test and Evaluation, Periodic Performance Testing, and Performance Confirmation), this plan only addresses the last three elements. Both site characterization and performance confirmation are currently being addressed in detail by other plans and/or activities. The sections that follow briefly discuss concepts for each of the remaining functional elements.

3.3.2.1 Developmental Test and Evaluation

Developmental Test and Evaluation includes development testing to support design and reduce development risks, and qualification testing to verify design compliance with requirements and specifications and to evaluate compliance with government regulations. Development testing is used to confirm design concepts, evaluate alternative design concepts, provide proof of concept documentation, and show the availability of needed technology. Development testing can involve prototype build and mock-up support. These tests are planned and conducted by the design organization as part of the Proof of Concept Design Process.

Waste package materials testing and modeling is being conducted during site characterization. This effort is divided into two major activities: waste package/engineered barrier system materials testing and modeling, and waste form materials testing and modeling. The main focus of the former work is on container materials, which is subdivided into five technical efforts: long-term corrosion, humid air corrosion, crack growth tests, electrochemical potential studies, and microbiologically influenced corrosion. Other efforts include testing of basket material corrosion and ceramic and other engineered barrier materials evolution. In addition, models are developed for each degradation mode to permit prediction of long-term performance. The waste form materials testing and modeling task includes testing of oxidation and flow-through release in spent fuel, unsaturated testing of both spent fuel and glass, and modeling of the behavior of both waste forms. Future efforts could include tests on DOE spent nuclear fuel and spent fuel cladding and hardware.

3.3.2.2 Operational Test and Evaluation

Operational Test and Evaluation tests and evaluates the operational suitability and effectiveness of the repository, its compliance with design/licensing-basis requirements, the product compliance with the operational requirements, and the repository's impact on the environment while operational. Operational Test and Evaluation includes integration and system testing and demonstration, beginning with the authorization to construct the repository and ending when all the Operational Test and Evaluation objectives have been met. These tests and demonstrations involve operational procedures and personnel. The startup test will be performed on the entire MGDS and will combine all the structures, systems, and components, utility systems, facilities, and processes required to receive, prepare, emplace, and retrieve waste. The startup test will be conducted by trained operators and performed using certified operational procedures. The final verification of the MGDS requirements is a major outcome of this test.

3.3.2.3 Periodic Performance Testing

Periodic Performance Testing will provide an ongoing verification that the operational systems within the MGDS continue to function in a safe manner, thereby ensuring safe handling of radioactive materials. These test requirements may be specified as part of the license to receive waste, 10 CFR 60, 10 CFR 20, other federal and state codes and regulations, manufacturer operations and maintenance manuals, or as an item of critical importance in a waste processing flow. A test working group will be appointed to plan and document the test requirements, including the development of the detailed test procedures for each unit under test. Selected test procedures will be demonstrated during the Startup Test. Implementation of the Periodic Performance Testing will commence with the successful completion of the Startup Test.

CHAPTER 4 – DESIGN AND CONSTRUCTION

The primary design emphasis this reporting period was on completion of activities to support the viability assessment and its key activities. The role of design is to provide a reference viability assessment design, to support the performance of testing and modeling required for the data input needs of the total system performance assessment, and to ensure that design information is sufficient to support the viability assessment cost analysis.

Progress was achieved in many design and construction activities. Work involved development of design processes, design concepts, and design guidance to support future design for the license application; identification of major design issues for resolution and reduction of uncertainties; and development of processes that support fabrication and testing of the waste package. Construction emphasis was on completion of the main tunnel loop and associated alcoves and on preparations for adding a full east-west traverse of the proposed repository block to support site characterization.

Key advances were made in the following activities:

Completion of a reference design to support the total system performance assessment – To accomplish this major project milestone, a reference design was selected as the basis for total system performance assessment design. The design provided the initial basis for the performance assessment calculations. Further development of design will continue, associated with the viability assessment, to provide quality information and to form the basis for the viability assessment cost. Design options will be addressed through sensitivity evaluations during the performance assessment.

Completion of the Exploratory Studies Facility (ESF) main tunnel loop – Another major project milestone, the ESF main loop, was completed in late April 1997. The loop is approximately five miles in length. One of the associated alcoves and two test niches were also completed.

Identifying structure, system, and component design bases through implementation of Compliance Program – This program, begun in this report period, will evaluate the applicability of codes, standards, and other guidance documents to the design, construction, and operation of the repository. The results will be input to System Descriptions Documents as design bases and will determine the applicability of industry guidance in non-engineering topics. Implementation of this program early in the design process will allow efficient use of time during the one-pass approach to design and will facilitate NRC review of project activities.

Identification of major design issues related to site recommendation and license application, and development of a plan for resolution – The major issues associated with design that have great impact on performance were identified (site recommendation/license application design ssues). A preliminary plan for resolution of these issues prior to licensing has been established. This plan will assure proper addressing and documentation of assumptions and uncertainties, improve quality of design, and reduce uncertainties in preparation for licensing.

Tentative resolutions have been selected for the viability assessment design issues, which provide interim positions as the bases for design and capture the major factors in selecting a repository design. Closure of these issues has been pursued during this period, and associated documentation is being developed.

Identification of viability assessment design options – Design options have been identified. Four options are being pursued in depth as part of the viability assessment; the remaining options will be pursued for the license application. The four viability assessment options include:

- Drip shields
- Ceramic coating
- Backfill
- Cladding credit.

Sensitivity analyses on each option will be performed as part of the viability assessment to establish the merit of each option, which will determine the options that should be further pursued.

Criticality Report Update – The *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997b) was updated during this period. The report describes the analysis methodology planned for use in demonstrating postclosure criticality control for the repository and captures the current status of the analysis methodology assumed in the waste package viability assessment design. The update reflected significant progress on analyses associated with pressurized water reactor fuel, burnup credit, and internal and external waste package criticality.

Development of alternative approaches to the design of the repository drift linings – Three alternatives were developed for the design approach to the repository drift linings. Eventual selection among the alternatives will be based on results from the total system performance assessment and the need for geologic mapping of the drifts. The three alternatives are precast concrete, cast-in-place concrete, and steel.

Selection of key design concepts for remote operations – Substantial progress was made in developing concepts for remote operations. Key reference viability assessment design concepts were identified and selected for waste package handling (emplacement, transport, and retrieval) equipment and for performance confirmation. Preliminary drawings, descriptions, and data sheets for this equipment were developed.

4.1 GENERAL

4.1.1 Design Bases

Emphasis has been placed on development of design bases early in the design process to ensure efficiency in development of a design for the license application. This work has progressed in the elements listed here.

Mined Geologic Disposal System (MGDS) Requirements – The *MGDS Requirements Document* (MGDS-RD) (DOE 1997a) contains the system-level requirements that the MGDS must meet to fulfill its mission to receive and dispose safely of high-level waste. Since the last reporting period, a Revision 3 draft was prepared. This draft document supports the requirement hierarchy streamlining concept initiated by the Civilian Radioactive Waste Management System (CRWMS) *Requirements Document*, Revision 3 (DOE 1996c). Also included are requirements for areal mass thermal loading and a period of retrievability that, for conservatism, is twice that mandated by 10 CFR 60.111(b)(1). The draft revision refines and simplifies presentation of the requirements. Formal issuance of this draft as Revision 3 will occur in fiscal year (FY) 1998.

Controlled Design Assumptions – The *Controlled Design Assumptions Document* (CRWMS M&O 1997d) was updated twice during this reporting period. This document contains requirements, data, key project assumptions, and regulatory interpretations that must be verified; these are captured to ensure this information is used consistently throughout the Project. The first update added two engineered barrier system performance goals, a requirement for the engineered barrier system to include multiple barriers, and more current data on seeping water chemistry and flow rates. To resolve issues related to the viability assessment, the second update added approximately 40 new assumptions, modified 23 assumptions, and withdrew 10 assumptions. A new appendix was added to describe each viability assessment key design issue, identify the resolution concept for each issue, and tie the applicable assumptions to each issue.

Compliance Program – The Compliance Program is a formal program that is under development to assess regulatory guidance documents and industry experience documents to determine applicability to design of the MGDS. This determination, along with supporting rationale, will be documented in Compliance Program guidance packages. Compliance Program guidance packages will serve as input to the system description documents upon which the MGDS design will be based. The information will also input into the MGDS license application. Conformance of the design with guidance provided in the Compliance Program guidance packages will ensure that the design is consistent with the applicable established regulatory precedent and will facilitate the NRC's timely review of the license application. Compliance Program guidance packages will also be used as input, where applicable, to engineering design guides, to design analyses related to design basis events, and eventually to the license application. The scope of the Compliance Program is being expanded in FY 1998 to include assessment of regulatory guidance documents and industry experience documents potentially applicable to non-engineering topics, such as emergency planning and radiological protection.

During this reporting period Compliance Program processes were developed and refined. Schedules and priorities were established, and a draft procedure (NLP-3-36) was prepared to govern implementation. Three prototype Compliance Program guidance packages were developed, which focused on identifying applicable guidance, documenting rationale, and providing justifications. A database was established to contain the information and to provide ready access to individuals who will use the data.

In FY 1998, Compliance Program guidance packages will be developed that address Bin 3 (systems related to nuclear safety or waste isolation, but have no regulatory precedent) structures, systems, and components. Some of the Bin 2 system description documents will also be developed, as will documents for many non-engineering Program topics. The rest of the packages will be prepared in the future.

System Description Documents – System Description Documents are under development that will contain design, construction, descriptive, and operational information for each of the systems that make up the MGDS. This period's emphasis has been on the development of Section 1, Functions and Criteria, of the System Description Documents. Of the 56 systems identified, Section 1 drafts for 28 System Description Documents were completed during this period. Development priority is assigned based on a previously developed binning system. Structures, systems, and components are prioritized on the basis of potential importance to radiological safety and/or waste isolation, as well as the availability of applicable licensing and design precedents. In addition, structures, systems, and components are binned based on regulatory significance of the structures, systems, and components in the licensing process (e.g., safeguards and security, fire hazards). The bins are defined as follows:

- Bin 1: Systems not related to nuclear safety or waste isolation (i.e., primarily balance of plant systems)

  Examples of Bin 1 Systems:
  - MGDS Site Layout
  - Site Compressed Air System
  - Subsurface Compressed Air System
  - Subsurface Water Collection/Removal System
  - Subsurface Water Distribution System

- Bin 2: Systems related to nuclear safety or waste isolation that have regulatory precedent

  Example of Bin 2 Systems:
  - Waste Handling Building Ventilation System
  - Waste Treatment Building Facility
  - Waste Handling Building Electrical System
  - Subsurface Electrical Distribution System
  - Site Communications System

- Bin 3: Systems related to nuclear safety or waste isolation that have no regulatory precedent

  Examples of Bin 3 systems:
  - Subsurface Ventilation System
  - Uncanistered SNF Disposal Container
  - Canister Transfer System

- Ground Control System
- Waste Emplacement System
- Assembly Transfer System

Section 1 drafts for all the identified Bin 3 systems, which are of the highest priority, were completed. Criteria in the System Description Documents come from a variety of sources (e.g., design analysis, studies, management decisions, and existing regulations). A primary source of regulatory guidance is established through the Compliance Program.

The plan for FY 1998 is to issue initial versions of Section 1 of all the Bin 3 System Description Documents and part of the Bin 2s.

4.1.2 Mined Geologic Disposal System Concept of Operations Document

The *Mined Geologic Disposal System Concept of Operations* document (CRWMS M&O 1997h) was revised and updated during this reporting period to reflect the viability assessment design and operating concept. Its objective is to ensure a common understanding of MGDS operations among system planners, developers, and implementers by integrating design solutions with operating concepts. This document provides an integrated, conceptual description of the physical architecture and operating concept of the MGDS.

Significant changes from the *Mined Geologic Disposal System Advanced Conceptual Design Report* (CRWMS M&O 1996b) to the viability assessment design that are now captured in the Concept of Operations document include the following:

- A description of the natural environment conditions at Yucca Mountain is presented with both current and predicted future trends, which include the most recent climate, surface hydrology, and seismic information being used by the Project.

- The types and quantities of waste that will be accepted by CRWMS for disposal are presented. These include commercial, U.S. Department of Energy (DOE) and U.S. Navy spent nuclear fuel, as well as high-level waste. Other waste forms are being evaluated for potential disposal in the repository. These include greater than class C low-level wastes, Cesium/Strontium capsules, immobilized plutonium waste forms, and DOE Special Case Waste.

- The north portal pad layout was modified to incorporate cask maintenance facility functions in the waste handling building, to provide for wet handling of bare spent nuclear fuel and dual purpose canisters, and to provide for the dry handling of high-level waste, DOE spent nuclear fuel, and commercial spent nuclear fuel in disposable containers.

- Two transportation formats are addressed for waste transportation within the State of Nevada: rail operations linked to heavy-haul trucks to the repository, and the addition of rail spurs to provide rail access to the repository.

- The subsurface layout was updated by expanding the upper block emplacement area (allowing the lower block to be classified as an expansion area), deleting the tunnel boring machine launch mains, and lowering the main ventilation exhaust drift below the plane of the repository. The operating concept of the subsurface operations was updated so that disposal container emplacement is performed with a gantry system that lifts the loaded disposal container from the subsurface transporter rail car. This system then places the disposal container onto the waste package support assembly at a preselected emplacement drift location.

- The strategy for assuring that public health and safety are protected after closure of a Yucca Mountain repository was updated. The approach has containers for high-level waste and DOE spent nuclear fuel being co-disposed between disposal containers containing commercial spent nuclear fuel. This approach includes a co-disposal container that holds five defense high-level waste canisters, arranged around a center-positioned DOE spent nuclear fuel canister. These containers contribute minimally to overall thermal loading but do reduce cold spots. This co-disposal container is emplaced between commercial spent nuclear fuel disposal containers that are spaced to provide an areal mass thermal loading of 85 MTU/acre. Separate disposal containers, loaded only with DOE spent nuclear fuel, are also assumed to be emplaced between commercial spent nuclear fuel disposal containers in the same manner.

4.1.3 Reference Design Description Document

A new procedure was developed to provide management control and guidance for the CRWMS Management and Operating Contractor (M&O). It describes the process for approval of initial design and design changes of the reference repository. Included in this procedure is the process controlling development of the *Reference Design Description for a Geologic Repository* (CRWMS M&O 1997i). This reference design description describes the current design expectations for a potential geologic repository, which could be located at Yucca Mountain. This description of the current repository design forms the basis for the cost estimate of the total system life-cycle cost that will be performed in FY 1998.

A reference design description (CRWMS M&O 1997i) was developed and revised during this reporting period. The current version is Revision 1. As a system integration and communication tool, the reference design description is consistent with the design for the viability assessment. It discusses design parameters, repository layout, ventilation, ground support, waste handling facilities and operations, disposal canister configurations, emplacement concepts and systems, caretaker operations, closure operations, and integration of the individual features of the engineered barrier system. The reference design description includes a description of the facilities and equipment required to support performance confirmation activities. Also described are optional engineered barrier system features, under investigation as part of the viability assessment design. (See Chapter 3, Figures 3-3 and 3-4.) These include cladding credit, a waste package ceramic coating, various waste package drip shields, and backfill to ensure that corrosion of the waste package is not enhanced and that the ceramic coating (if used) is protected. A conceptual description of the backfill emplacement concept is provided.

4.1.4 Cost Analysis Report

The 1997 *Cost Analysis Report* (CRWMS M&O 1997a) was developed that will provide guidance for preparation of the 1998 viability assessment cost estimate. It presented the assumptions and format envisioned for that cost estimate, as well as a current project estimate based on the current reference design. The cost estimate covers the period beginning with submittal of a license application; it reflects the cost to complete the repository and engineered barrier designs, to construct and operate the repository, and to close and decommission the repository. The MGDS viability assessment cost estimate will be prepared during FY 1998.

4.1.5 Design Issues

During this period, viability assessment key design issues were addressed, and site recommendations and license applications design issues were selected, developed, and documented. A database was established to contain information regarding identification of each issue and progress toward resolution. A preliminary plan for resolution of each issue has been established. This plan will ensure that assumptions and uncertainties are properly addressed and documented, improve the quality of design, and reduce uncertainties in preparation for licensing.

Viability Assessment Issues – The 20 issues identified as Key Design Issues in Progress Report #16 (DOE 1997b) were selected as the design issues deemed important for viability assessment reference design. The Key Design Issues identified for viability assessment include:

- Thermal loading range
- Criticality control
- Performance confirmation concept
- Confirmation of high volume and long period waste handling capability and design basis event consequences
- Strategy for mapping repository subsurface
- Viability of underground remote control concepts
- Regional Service Contractors/Interim Storage Facility interface
- Waste package sizes and weights
- Design basis model
- Surface development
- Engineered barrier system performance enhancements

- Emplacement drift ground support concept
- Retrievability concept
- Disposal of site-generated waste
- Postclosure performance standards
- Repository seals requirements and concepts
- Additional waste forms
- Waste package materials
- Subsurface development
- Site development.

Work continued on the tasks necessary to resolve these 20 viability assessment issues. None of the issues have been resolved sufficiently for closure. All issues are scheduled to be resolved to the extent necessary to support the viability assessment between January 30 and May 30, 1998.

The *VA Design Issues Status Report* (CRWMS M&O 1997j) was prepared to present the status of progress on the viability assessment issues at the end of FY 1997.

Site Recommendation and License Application Design Issues – During this reporting period more than 100 design topics were reviewed, and 13 were selected as site recommendation and license application design issues because of their significant bearing on the site recommendation and/or the license application. Some of the site recommendation and license application design issues are a continuation of the viability assessment design issues, reflecting additional work needed to address these issues for a site recommendation and license application. The titles of the issues identified include:

- Thermal Loading Reference Values
- Engineered Barrier System Performance Enhancements and Reliability Requirements
- Description for Supportable Design and Licensing Basis of Flux
- Ground Surface Temperature Limits
- MGDS Interface with Regional Service Contractors and/or Interim Storage Facility
- Waste Package Materials Selection for the License Application
- Defense-in-Depth Design Basis
- Waste Package Emplacement for Thermal Management
- Criticality Control Concepts; Burnup Credit Limits
- Postclosure Performance Standards
- Near-Field Environment Design Basis

- Additional Waste Forms
- Acquisition of Qualified Technical Data Base.

4.1.6 Environmental Impact Statement Support

Repository Design provided design data needed by the environmental impact statement contractor to prepare the environmental impact statement and to evaluate options and alternatives identified in the Notice of Intent. The data included descriptions of the program phases, design evolution, and repository designs. Summary level engineering values needed by the environmental impact statement contractor (i.e., staffing, wastes, emissions, resources, and land use) were also included. Environmental data were developed using underground opening layouts appropriate to the thermal load and waste inventory cases.

In FY 1998, engineering data will be updated to include additional cases of interest to the environmental impact statement contractor. These cases include designs that may accommodate potentially larger and more varied waste inventories, an operational waste storage facility, and an intermodal waste transfer facility (rail to heavy-haul).

4.2 REPOSITORY DESIGN

Repository design progress includes work on subsurface and surface design for the MGDS. An integral part of repository work includes interfacing with waste package design. The primary focus during this period was on work toward providing the design to support the total system performance assessment-viability assessment (TSPA-VA), and toward providing a reference design for viability assessment.

4.2.1 Subsurface Design

4.2.1.1 Radiation Protection Design

Radiation Shielding Design Guide – A radiation shielding design guide (CRWMS M&O 1997l) was developed to document the approach to be used in the radiological design of facilities. This guide provides reference information such as radiation source data, material data, design criteria, and assumptions for use in preliminary repository shielding design analysis. The guide also identifies the selected computer codes, which include MCNP4A, SCALE4.3, QAD-CGGP, PATH, and MICROSHIELD.

Repository Radiological Design – To support the development of a viability assessment design, parameters for the source term for determining radiation fields needed to be identified to support shielding design of the waste package transporter. The transporter's design, in turn, was needed for design of the transport rail system between the surface facility to the subsurface emplacement drifts. Initial shielding analysis was performed for waste packages containing pressurized water reactor design basis fuel with the characteristics of 4.2 percent initial enrichment, 48.1 GWd/MTU burnup, and 10 years cooling. This design basis fuel results in conservative radiation fields that are higher than those of the average container of spent nuclear fuel by a factor of about five (CRWMS M&O 1997l).

The use of a single concrete shadow shield at the entrance to each emplacement drift was identified as an effective means of limiting the combined direct and scattered radiation field in the occupied main drifts to less than 0.8 mrem/hr (CRWMS M&O 1997m, Section 8, pg. 102). The worker occupancy classification of the main drifts can be changed from intermittent, without the shadow shields, to normal, with shields.

In FY 1998, preliminary scattered radiation modeling analysis will be performed as related to the design of emplacement drift shadow shields. Limiting specific source terms for radiation shielding of waste packages containing commercial spent nuclear fuel will be developed.

Radiological Aspects of Transporter Design – In support of the viability assessment design, a preliminary shielding design for the transporter was identified based on containers holding the design basis fuel gamma and neutron radiation source and worst-case external contributions from the respective radiation fields. The design consists of approximately 15.2 cm of carbon steel and 10.2 cm of 1.5 percent borated polyethylene for the radial component with similar dimensions for the transporter ends. The design results in a radiation field of 33 mrem/hr at the surface of the transporter (CRWMS M&O 1997m, Section 7.5.4, p. 84).

Alternative materials for transporter neutron shielding will be evaluated in FY 1998.

Radiological Considerations for Waste Package Retrieval – The viability assessment design required an evaluation of the retrieval of intact and breached waste packages. For normal retrieval of intact waste packages, the radiation field in the area of the main drift was calculated as varying from 1 to 100 mrem/hr (CRWMS M&O 1997n, Section 7.2, p. 50). Shielding requirements were identified for special equipment that may be required to retrieve waste packages during abnormal conditions (CRWMS M&O 1997n, Section 7.3, p. 60). These requirements consider the need for personnel to operate some special equipment locally. Potential airborne radiation hazards for retrieval of breached waste packages were also identified.

A methodology will be developed in FY 1998 to assist in determining the transport of radioactive materials from a breached waste package and the subsequent contamination of repository surfaces.

4.2.1.2 Waste Package Handling and Equipment

Major Equipment – In support of the viability assessment design, conceptual equipment and a viable handling concept were developed for the receipt, packaging, transport, and emplacement of various sizes of waste packages containing high-level radioactive waste. While all the recommended mobile equipment will be both first-of-a-kind and one-of-a-kind for this application, the basic concept of rail transportation and the handling of radioactive materials in a shielded, remotely controlled environment has been based on reliable, proven technology with reasonable relative costs. This work has been documented in two related design analyses (CRWMS M&O 1997o; 1997p).

Conceptually, the waste emplacement system will transport the sealed and loaded disposal container from the surface waste handling building to the area of emplacement. This system operates on the surface between the waste handling building and the north portal and in the underground ramp, mains, accesses, and emplacement drifts. The system accepts the disposal containers, preloaded into a shielded transporter, transports the disposal container to the emplacement area, and emplaces the disposal container in the emplacement drift (CRWMS M&O 1997o, pp. 16-17).

Major transport equipment includes a shielded transporter with an integral waste package loading/unloading mechanism. Transport locomotives include features for both local and remote control operation. Major emplacement equipment includes a remotely controlled gantry for waste package emplacement in an emplacement drift and a gantry carrier for transfer of the gantry between various emplacement drifts (CRWMS M&O 1997o, p. 8). The 5.5-m diameter of the emplacement drifts was developed from the operating envelope of the waste emplacement equipment required to handle the largest waste packages. The size of the drifts is an input to the total system performance assessment and to the design and cost elements of the viability assessment.

The *Repository Rail Electrification Analysis* (CRWMS M&O 1997p, pp. 61-62) conceptually selected a 750 VDC electric power supply for mobile equipment traveling between the waste handling building and the subsurface repository and within the emplacement drifts. An electrified third rail will serve as the electric conductor in emplacement drifts, and a single overhead trolley wire will serve in all other subsurface openings. These are commonly used systems and were selected after considering function, worker safety, and cost.

Work on the waste handling equipment during FY 1998 will consist of structural evaluations and further development of mechanical details, with focus on the waste package transporter, gantry carrier, and reusable rail car.

Remote Operations – Based on a series of preliminary design analyses and a review of available remote control technologies, several key reference viability assessment design concepts were identified for remote operations equipment for waste package handling (emplacement and retrieval) and for performance confirmation. Remote operation and control of key subsurface waste handling and repository monitoring activities are required because of the high radiation fields and elevated thermal conditions in the vicinity of the waste packages. Individual reference concepts were selected from among several alternatives based on factors such as functionality, personnel safety, reliability, long-term maintenance requirements and maintainability, life-cycle costs, experience with the technology, commercial availability, survivability, complexity, and degree of adaptation that may be required.

Design analyses during this period examined emplacement equipment locomotion and mechanical actuation, vehicle electrification, and aspects of the waste emplacement system control and communication systems (CRWMS M&O 1997o; 1997p; 1997q; 1997r). Methods were identified for monitoring and controlling the operation of mobile equipment such as the emplacement gantry, emplacement locomotives, waste package transporter, and gantry carrier.

Preliminary design analysis investigated implementation of a vehicle control system based on dual redundant programmable-logic-controllers, which integrate control of vehicle locomotion, braking, power supply and distribution, wireless communication, vision systems, thermal monitoring and control, radiological monitoring, mechanical actuator control, and safety and auxiliary systems (CRWMS M&O 1997r, pp. 65-66). Initial sizing of the emplacement system control system, including remote control of the drift isolation doors and rail switches, was performed. Design strategies for achieving high system reliability and availability were considered. These strategies included adherence to appropriate quality assurance design practices, use of redundant components, use of diverse technologies, physical separation of primary and backup systems, and incorporation of fault-tolerant design features.

Preliminary design analyses also developed reference designs for remotely operated waste package retrieval equipment (CRWMS M&O 1997s). Alternative concepts for abnormal retrieval scenarios included remote excavation and hauling, as well as portable shielding and shielded equipment that allow for local operation.

Two design analyses were completed to establish reference viability assessment concepts that support the performance confirmation program (CRWMS M&O 1997t; 1997u). A data acquisition system was developed for long-term monitoring and evaluation of the repository. Remote operations design activities supported the viability assessment design and the viability assessment cost estimate. Facility considerations for performance confirmation are described in Section 4.2.1.5.

Activities during FY 1998 will include development of an overall integrated control plan for the repository and further development of the data communication concepts for important-to-safety systems.

Retrieval – In support of the viability assessment design, a viable equipment concept was developed for waste package retrieval under both normal and abnormal conditions. This work was documented in a design analysis, *Waste Package Retrieval Equipment* (CRWMS M&O 1997s). Waste package retrieval under normal conditions would be accomplished with the same fleet of equipment that would be used for waste package emplacement. The abnormal retrieval process and equipment would be used when conditions preclude the use of the normal retrieval process and equipment. Abnormal retrieval situations are based on events that are described in the *Preliminary MGDS Hazards Analysis* (CRWMS M&O 1996a).

The following is a preliminary list of major equipment (CRWMS M&O 1996a, Section 7.2.3.1, p. 51) that would be needed under various abnormal retrieval scenarios. Each piece of equipment can be developed using proven technology.

- Heavy Duty Forklift
- Inclined Plane Hauler
- Bottom Lift Transporter
- Multipurpose Vehicle
- Load Haul Dump Loader
- Covered Shuttle Car
- Scaler.

The sequence of the normal retrieval operation is the reverse of the emplacement cycle, and the same equipment can be used. The equipment and concept for abnormal retrieval conditions are based on preliminary information; additional development work will be required.

In FY 1998 details of waste package handling systems for normal retrieval will be refined.

4.2.1.3 Thermal Management

Results of the total system performance assessment depend strongly on the thermal loading strategy of the repository as well as the pattern(s) selected for emplacement of waste packages. Repository thermal loading refers to the density at which waste packages will be emplaced in the repository. The thermal loading strategy significantly affects design and layout of the emplacement drifts within the repository, and therefore the viability assessment design. It also substantially affects the viability assessment cost estimate.

Two different loading strategies have been extensively examined: line loading and point loading. Line loading refers to a waste package arrangement that is essentially end-to-end, with little or no space between packages. Point loading refers to an arrangement in which some nominal space exists between waste packages. Line loading provides for an improved local temperature profile, but the density results in potential problems with exceeding thermal limits of the drift wall. While line loading is currently identified as a design option, reference design for total system performance assessment is point loading of the waste packages. As a part of the overall thermal loading studies completed in support of the viability assessment, the following subjects were evaluated during this reporting period: maximum thermal loading to meet temperature goals at the top of the zeolite layer, emplacement drift and waste package spacing, emplacement strategies, and emplacement schedule.

A geochemical thermal goal specifies that the temperature at the average top of the zeolite layer, which is 170 m beneath the potential emplacement area, shall not exceed 90°C (CRWMS M&O 1997d, DCSS 025). To meet the geochemical thermal goal, and considering modeling conservatism, an areal mass loading of 85 MTU/acre was recommended for purposes of the viability assessment design (CRWMS M&O 1997v, p. 23).

A series of emplacement drift spacing and waste package center-to-center spacing combinations were evaluated (CRWMS M&O 1997v, pp. 25-40; CRWMS M&O 1997w). Waste package physical dimensions and near-field thermal goals were considered (CRWMS M&O 1997d, DCSS 023). The goal of this study was to determine the minimum total length of the emplacement drifts necessary to meet all criteria. An emplacement drift spacing of 28 m was selected for the viability assessment design. The waste package spacings for the recommended thermal load and the selected drift spacing were established using the assumed waste stream identified in the *Controlled Design Assumptions Document*, Revision 4 (CRWMS M&O 1997d, Keys 003 and 004). The DOE high-level waste packages, which range in length from 3.70 to 5.40 m, will be emplaced in the spaces between the 21-pressurized water reactor and 44-boiling water reactor waste packages.

Selected waste emplacement configurations were evaluated in the *Determination of Waste Package Design Configurations* (CRWMS M&O 1997x). The analysis results indicated that certain placement configurations should be avoided to prevent approaching or exceeding thermal limits. For example, certain waste packages would contain a much greater-than-average heat output: up to the design basis heat output of approximately 18 kW per package (CRWMS M&O 1997x, p. 29). It was determined that, if possible, those waste packages that contain a much greater-than-average heat output limit should not be placed adjacent to each other. Certain combinations were not included in the analysis but will be evaluated in future design analyses. Restrictions may mean that it will not always be possible to emplace waste packages in a single emplacement drift in the order available from the waste handling building.

Evaluations and comparisons were performed for three emplacement approaches: areal power density (typical units of kW/area), areal mass load (typical units of MTU/area), and equivalent energy density (typical units of GJ/area). The thermal modeling results indicate that no emplacement strategy is capable of generating uniform rock temperatures in both the near term and long term. The areal mass load approach will produce temperature uniformity in the long term, while the areal power density approach creates reasonably even temperature distribution in the near term. Use of the equivalent energy density approach appears to result in a compromise between the areal power density and areal mass load approaches. With the equivalent energy density approach, each waste package would have a unique spacing depending on its characteristics. The results illustrate that the equivalent energy density approach may provide better preclosure management of near-field rock temperatures than the areal mass load approach and better postclosure management than the areal power density approach. Further evaluation is recommended in future thermal management studies (CRWMS M&O 1997v, p. 57).

A preliminary waste package emplacement plan by year was developed for the waste stream evaluated (CRWMS M&O 1997y). This scheme shows the progress of a 24-year waste emplacement period, starting in 2010 and ending in 2033, on a repository layout. The sequence and timing of waste package emplacement, in part, establish the initial conditions from which the total system performance assessment will proceed.

Thermal management work in FY 1998 will address a number of waste package emplacement issues relative to meeting the performance goals of the repository. These include:

- Waste stream variability
- Emplacement arrangement
- Areal mass loading versus areal power density versus equivalent energy density
- Disposition of non-commercial waste
- Issues related to thermal management strategy, such as edge loading, waste package relocation, surface and subsurface lag storage, and continuous ventilation of emplacement drifts.

Based on the results, the waste package emplacement strategy will be revised accordingly.

4.2.1.4 Seals and Closure

During this reporting period, the role of the sealing subsystem (shaft, ramp, and exploratory borehole seals) in achieving the overall performance objectives for the waste isolation system was evaluated in support of the total system performance assessment. Sealing is defined as the permanent closure of the shafts, ramps, and exploratory boreholes. Sealing includes those components that would reduce potential inflows above the repository, or that would divert flow near the repository horizon to allow vertical infiltration to flow around the repository. The sealing of emplacement drifts was not evaluated because the current capability to calculate fracture flow into the drifts is not yet sufficiently mature.

The *Repository Seals Requirements Study* (CRWMS M&O 1997z) provided water or air flow performance-based requirements for shafts, ramps, and exploratory boreholes located near the repository. These requirements will be placed in the Functions and Criteria section of systems description documents. Recommendations, as appropriate, were provided for developing plans, seals component testing, and other studies relating to sealing. The effort determined that it would be most appropriate to use subsystems standards as defined by the NRC in 10 CFR Part 60 and Nevada state regulatory requirements (NAC 534A.150, NAC 534A.490, and NAC 534A.500) [Nevada Administrative Code (NAC 534A), Geothermal Resources] in developing water flow or air flow performance-based criteria, which formed the bases of the sealing requirements developed in the study.

The study also included an assessment of the need for sealing shafts and ramps, which was performed by comparing the allowable water-flow goals to the anticipated water flows of the underground repository. The comparisons showed that seals would not be necessary for the anticipated water flows from the shafts, ramps, and the underground facility resulting from an infiltration rate of 5 mm/yr. At an infiltration rate of 30 mm/yr, the allowable flow rates would be exceeded and seals would be required. For unanticipated flows, flows through the shafts and ramps might also exceed the allowable water flows over short periods of time. For deep exploratory boreholes that intersect or go below the repository horizon, the need for repository seals is established on the basis that they could represent preferential pathways from the repository horizon to the groundwater table.

A performance allocation evaluated the relative ease or difficulty in sealing to establish the sealing criteria for each component. The major components considered were the seals in the shafts, ramps, and boreholes and the underground water storage capacity (the amount of temporary water storage volume at an elevation below the waste packages that allows holdup of water that will eventually drain into the rock and is large enough so that water will not contact the waste package containers). The performance allocation provided for the performance of each of these components.

Design relations were developed to provide a flexible approach for determining design flow rates through the shaft and ramp seal subsystem, and the total storage capacities at the base of the repository that meet the NRC requirements. On the basis of this work, a set of recommendations for sealing requirements was proposed. A brief synopsis of the recommendations for sealing requirements was that the subsurface design must provide for underground water storage, which will accommodate an inflow into the repository of 17,000 to 49,000 m^3/yr. The boreholes, shafts, and ramps will be sealed to limit the flow rate through those seals to values equal to or less than the value computed from the following equation:

$$\text{Flow rate} = (0.98 \bullet \text{Total Storage Capacity in } m^3/yr) - 13{,}234 \ m^3/yr$$

This flow rate can be achieved by providing seals having conductivities of 10^{-5} to 10^{-6} cm/s, with the higher conductivity corresponding to the higher flow. Additional details of the requirements can be found in the *Repository Seals Requirement Study* (CRWMS M&O 1997z).

If, in the regulatory process, a decision is reached to make the borehole or shaft and ramp seals more robust, the design can easily be changed by scaling the work done in this report. Lower conductivity seal material, backfill, or larger seal plugs could all be considered to increase the robustness of the seal.

Design guidance for the environmental conditions and the lifetime that the seals design must meet were also provided in the analysis.

In FY 1998, additional evaluations of possible flow into the emplacement drifts will be performed to evaluate the performance impact of engineered water drainage that may concentrate radionuclides in a localized area within the waste emplacement drift. The capability to perform such evaluations should be available in spring 1998.

4.2.1.5 Performance Confirmation Design Considerations

Performance confirmation draws its requirements from 10 CFR 60, Subpart F and uses the total system performance assessment for information related to expected repository performance. In some cases, there is an iterative relationship between performance confirmation and preferred repository layout, with subsequent impact on cost. During this reporting period, design considerations consisted of first determining the types of data that must be obtained, then conceptually developing the methods and subsurface facilities required to obtain the data. Performance confirmation parameters, including types of data required, were obtained from the *Performance Confirmation Concepts Study Report* (CRWMS M&O 1997aa), which was developed to meet the requirements of 10 CFR 60, Subpart F.

The design analysis, *Subsurface Repository Performance Confirmation Facilities* (CRWMS M&O 1997ab), determined both the general methodology for acquiring performance confirmation data and requirements of the subsurface performance confirmation facilities. This analysis described how these would be incorporated into the overall repository design. The analysis also evaluated performance confirmation parameters that may lead to the retrieval of waste packages.

The analysis developed the borehole drilling approaches for obtaining data and identified data that could be obtained by testing dummy waste packages. The analysis developed configurations for the performance confirmation testing facilities, including performance confirmation drifts above the emplacement drifts horizon and testing alcoves excavated horizontally from the access mains. The analysis examined access and ventilation needs and methods for constructing the facilities. It concluded that the tunnel boring machine that was used for excavating emplacement drifts could also be used for excavating performance confirmation drifts. The analysis also examined the thermal impact of the waste packages on the performance confirmation drifts, and concluded that performance confirmation drifts located as close as 10 m above the emplacement would be viable with adequate ventilation (CRWMS M&O 1997ab, p. 52).

The design analysis, *Performance Confirmation Data Acquisition System* (CRWMS M&O 1997u), examined data acquisition systems and how these interfaced with the subsurface facilities. Section 4.2.1.2 presents information regarding the monitoring systems.

During FY 1998, systems engineering studies will be undertaken to refine the performance confirmation parameters. This information will form the basis for revisions to the current performance confirmation facilities and data acquisition designs. Work will also re-examine the means for acquiring the data, including borehole instrumentation and in-drift monitoring systems.

4.2.1.6 Ventilation Systems

The overall subsurface ventilation system design concept was updated to be compatible with the current viability assessment (CRWMS M&O 1997i) repository layout. Because air-flow velocities, volumes, and flow paths contribute to the removal of heat and moisture, these are inputs to the total system performance assessment. Viability assessment design and viability assessment cost estimates were also supported. All findings and progress included in this report are found in *Overall Development and Emplacement Ventilation Systems* (CRWMS M&O 1997ac).

Air Velocities – Minimum drift air velocities were established for the various locations and times during the life of the repository. From such data, the total repository airflow quantities during construction, operations, and caretaker phases were established (CRWMS M&O 1997ac, p. 40). The volumes established were within the range commonly used for large mining operations.

Air-flow velocities will be further studied during FY 1998 as part of the ventilation studies described in "Ventilation Modes" in this sub-section.

Dust Control – The project approach to dust control was established. The potential repository horizon contains levels of silica and cristobalite that require attention in the design, construction, and operation of the facility. A diversified approach is envisioned, which incorporates a suite of varied dust control measures. Effective, careful use of water is a historically proven dust control measure. Main sources of dust production, including excavation operations, excavated rock handling and dumping, and conveyor belt operations, will incorporate the use of water to allay dust. Because of low allowable breathable dust standards, additional measures are planned. These may include free-standing fan/filtration units to clean dust from the air stream, rigid "housekeeping" standards to reduce dust re-entrainment, training of subsurface personnel in the hazards of breathable silica and the importance of dust control, and incentive plans to promote and reinforce the dust control objectives. In addition to these measures, the development ventilation system was re-configured to make the south ramp the main exhaust. This places the conveyor, also in the south ramp, in the exhaust air stream and minimizes the contribution of dust by the conveyor to the development air stream (CRWMS M&O 1997ac, pp. 52-75).

The dust control program will be further developed during FY 1998.

Ventilation Modes – The ventilation system viability assessment design was completed, providing a basis for viability assessment cost estimating. Ventilation modes for the repository construction, operation, and caretaker phases were established. The intake and return airways will use the full drift openings (flow-through ventilation) to minimize energy requirements and maintenance (CRWMS M&O 1997ac, pp. 62-93).

The site ventilation system design consists of separate ventilation systems for the development side and emplacement sides of the repository. Ventilation barriers will be arranged to separate the development and emplacement side ventilation systems as required by 10 CFR 60.133(g)(3). The development side will have a positive pressure, and the emplacement side will have a negative pressure. This pressure difference will ensure that any leakage will always be from the development side to the emplacement side of the repository, so that workers on the development side will not be exposed to any radioactive releases, and that any radioactive releases will remain on the emplacement side where they can be captured (CRWMS M&O 1997ac, p. 43).

The overall ventilation system design addresses the control of potential exposure of personnel to heat and radionuclide releases. The return air from emplacement drifts will pass through an insulated metal duct to isolate it from human contact. The duct will lead into a subsurface high-efficiency particulate air filtration system, with high-efficiency particulate air filters on standby mode and automatically activated when needed (CRWMS M&O 1997ac, pp. 32-34).

A preliminary design for the construction ventilation system will be developed during FY 1998. The ventilation needs of the emplacement drifts will be studied further to include control mechanisms, high-efficiency particulate air filtration, instrumentation and monitoring during operation, backfill, and retrieval.

4.2.1.7 Ground Support

In this reporting period, three design approaches for ground support were developed to address the various potential constraints. These alternative approaches use precast concrete, cast-in-place concrete, and steel. If the total system performance assessment shows that postclosure interactions between concrete and rock do not meet the performance requirements, then the steel alternative will be selected. For drifts where geologic mapping is established as a requirement (see Chapter 3, Section 3.3.1), the precast concrete ground support system alternative is eliminated because it precludes subsequent mapping. The design for emplacement drift ground support is important as an input to the TSPA-VA and is part of the viability assessment design and the cost estimating.

Design Results – Thermomechanical analysis of permanent ground support for emplacement drifts has included examination of concrete and steel lining systems. For a concrete lining bonded to the rock, analysis using a model that accounted for concrete creep showed that computed lining strain levels were satisfactory compared with an ultimate concrete compressive strain limit of 0.3 percent. Ultimate strains were used in the model, because existing concrete codes do not cover material behavior at the elevated repository temperatures. Further testing and analysis are being initiated to establish the appropriate allowable stresses and strains. Ratios of ultimate to computed strains ranged from 1.8 to 2.1. Analysis of emplacement drifts supported with steel ring beams (steel sets) showed that modeling a weak bond between the steel and rock significantly reduced the computed thermally-induced lining stress. This work is described in *Repository Ground Support Analysis for Viability Assessment* (CRWMS M&O 1997ad, pp. 55-56). This alternative is being carried as a viability assessment design issue, "Emplacement Drift Ground Support Concept." To model the effects of elevated temperature, the temperature-dependent properties of materials were selected for analysis. An additional analysis (CRWMS M&O 1997ae, pp. 48-57, 71-72) was performed to establish approaches to concrete mix designs to enhance durability and minimize the need for maintenance.

An analysis, *Constructability Considerations for Repository Drifts for Viability Assessment* (CRWMS M&O 1997af, pp. 45-53, 61-73, 89-95), was performed to describe the methods of lining fabrication and installation. The analysis also considered methods of construction and lining installation to accommodate the geologic mapping requirements for performance confirmation. Analysis will continue during FY 1998 on the design of joints in precast segmental linings. Compressible elements in the steel and concrete lining systems will be examined as an additional means of reducing thermal lining stresses.

The *Drift Ground Support Design Guide* (CRWMS M&O 1997ag), which was a first step to form the basis for a guide for the license application design, includes methodology for designing initial and final ground support for emplacement drifts and other openings in the immediate area of the repository. FY 1998 revisions to this guide will include further development of allowable load limits for evaluating the computed lining stresses and strains.

Work for FY 1998 includes incorporation of updated and qualified data in the thermal-mechanical models and continuation of the structural design of a precast concrete lining for the emplacement drift. A steel lining system will also continue to be developed as an optional ground support system. The effects of heating and cooling cycles on both the jointed rock mass and on the lining systems will be examined.

Material Selection – An analysis, *Materials for Emplacement Drift Ground Support* (CRWMS M&O 1997ae, Table 7.5), of concrete and steel materials for ground support suggested a candidate concrete mix design for an emplacement drift permanent lining, which uses Type V cement. Constituents of concrete and steel are described; the analysis results provided information for the total system performance assessment.

Concrete chemical tests were initiated at the Pennsylvania State University Materials Research Laboratory to aid in judging the postclosure effects of organic admixtures in concretes. Preliminary results show the production of excess water but no carbon dioxide gas. While these tests continue, plans are being made for mechanical testing of concrete at elevated temperatures. These tests will measure concrete pH and provide data with which pH changes may be predicted. The long-term pH of the concrete-groundwater system may affect waste isolation characteristics of the drift environment.

The scope of work for FY 1998 includes further analysis and testing on ground support material longevity, especially on assessment of the potential for concrete deterioration and steel corrosion.

4.2.1.8 Repository Layout

The volume of rock mass available for siting the subsurface repository was identified and defined based on accepted site data. The geology model was developed from data entered into the LYNX system database (CRWMS M&O 1997ah, p. 16). Using the volume of rock mass available, a repository subsurface layout was developed for the viability assessment to allow emplacement of at least 70,000 MTU of waste with enough flexibility to accommodate potential changes in site conditions (10 CFR 60.133(b)) or programmatic requirements. The layout incorporates the ESF openings and establishes the layout and geometry of the repository openings (CRWMS M&O 1997ai, Figure 7-1). The geologic units and repository layout affect the total system performance assessment, the viability assessment design, and the viability assessment cost estimate.

The volume of rock mass available for siting the subsurface repository will be updated during FY 1998 as additional site characterization data become available.

4.2.2 Surface Design

Emphasis during this reporting period was on completion of the analyses required to establish feasible system configurations and to provide preliminary site and facility layouts sufficient to support the viability assessment cost estimate.

4.2.2.1 Waste Handling Systems

The waste handling systems include the seven primary mechanical systems associated with spent nuclear fuel and high-level waste handling within the waste handling building. These systems include the carrier/cask transportation system, carrier/cask handling system, assembly transfer system, canister transfer system, disposal container handling system, carrier preparation building handling system, and waste package remediation system. The design uses the systems established in the *Waste Handling Systems Configuration Analysis* (CRWMS M&O 1997aj), and the *Waste Handling Mechanical Flow Diagrams* (CRWMS M&O 1997ak). Both were completed during the previous reporting period, and the *Surface Nuclear Facilities Space Program Analysis* (CRWMS M&O 1997al) was completed during this reporting period.

The *Canister Transfer System Analysis* (CRWMS M&O 1997am) was prepared to establish a conceptual system configuration of the structures, systems, and components in the waste handling building that are required to remove shipping canisters (large multipurpose canisters, small multipurpose canisters, defense high-level waste canisters, and DOE spent nuclear fuel canisters) from their transportation casks, transfer them into disposal containers (or into staging racks for future transfer), and transfer them to the disposal container handling system. The space requirements for major components were then determined as a basis for an overall space allocation of the two primary areas (i.e., the cask preparation/decontamination area and the canister handling fuel transfer cell). Similar analyses were begun for the disposal container handling system, the carrier/cask handling system, and the assembly transfer system.

The *Waste Handling Facilities Recovery Analysis* (CRWMS M&O 1997an) was performed to identify and evaluate operational failures in the waste handling surface facilities and to recommend procedures and equipment (both general and specialized) needed for recovery operations.

The following work is planned for FY 1998:

- Complete the following system configuration analyses: Disposal Container Handling System Analysis, the Carrier/Cask Handling System Analysis, and the Assembly Transfer System Analysis

- Prepare the Waste Package Remediation System Analysis, the Carrier/Cask Transportation System Analysis, and the Carrier Preparation Building Handling System Analysis

- Prepare component/issue analyses that address major components and driving issues associated with the Waste Handling Systems for the eight primary mechanical systems (e.g., Spent Nuclear Fuel Drying Component Analysis, Cask Cooldown Component Analysis, Pool System/P&ID Process Analysis).

4.2.2.2 Waste Treatment System

The waste treatment system includes the primary process systems associated with handling the site-generated radiological waste within the waste treatment building. The subsystems include the mixed waste transfer subsystem, the solid low-level waste processing subsystem, and the liquid low-level waste processing subsystem.

The *Secondary Waste Treatment Analysis* (CRWMS M&O 1997ao) was prepared to update the secondary waste treatment system design, and to reflect evolution of the waste handling systems and facilities design. It includes assessing anticipated secondary waste volume generation rates, establishing a low-level waste handling capacity in the waste treatment building, developing data to support the process flow diagrams for the waste treatment system, and preparing a material balance table that outlines the flow and conditions of major process streams for the waste treatment building.

The *Secondary Waste Treatment Flow Diagrams* (CRWMS M&O 1997ap) were prepared to show the major equipment; the sequence of movement and throughput rates for solid low-level waste processing; and the major equipment, piping and controls, material balance, piping material selection, and major line sizes for liquid low-level waste processing.

FY 1998 will see preparation of component/issue analyses that address major components and driving issues associated with the Waste Treatment System (e.g., *Chemical Liquid LLW/P&ID Analysis* and *LLW Equipment Analysis*).

4.2.2.3 Nuclear Facilities

The *Surface Nuclear Facilities Space Program Analysis* (CRWMS M&O 1997al) evaluated space requirements and the functional space relationships among the three surface nuclear buildings (waste handling building, waste treatment building, and carrier preparation building). Consideration was given to the primary areas, primary support areas, facility support areas, and miscellaneous building support areas. A set of spatial parameters (e.g., minimum square footage, minimum heights) and layout requirements (e.g., adjacency requirements, access requirements) was established to develop preliminary building layouts as general arrangement sketches. Attachment I of the analysis included the layouts as general arrangement figures.

The *Surface Nuclear Facilities HVAC Analysis* (CRWMS M&O 1997aq) provided an assessment of the ventilation air-flow requirements needed to maintain proper indoor air quality and comfort, and to establish confinement zones for the three surface nuclear buildings (waste handling building, waste treatment building, and carrier preparation building). This assessment was used to develop preliminary sizing of the major heating ventilation and air conditioning components and to develop air-flow diagrams. The air-flow diagrams are included in the reference analysis as heating ventilation and air conditioning air-flow diagrams in Attachment IV (for the waste handling building), Attachment V (for the waste treatment building), and Attachment VI (for the carrier preparation building).

The *Waste Handling Operations - Dose Assessment* (CRWMS M&O 1997ar) assessed the worker dose for normal and off-normal handling operations in the waste handing building and carrier preparation building and identified operations that could be modified to reduce occupational radiation exposure. Exposures were calculated from the time of arrival on site until the exit of the waste package from the waste handling building.

The scope of the repository waste handling operations simulation model, which uses the WITNESS simulation modeling program, was expanded to cover both surface and subsurface waste handling operations (i.e., waste receipt and inspection, truck and rail parking, carrier staging shed operations, underground waste haulage, and in-drift emplacement). This integrated model can be used to determine throughput capabilities for various "what-if" options to the current design basis.

The *Remote Operations Design Guide* (CRWMS M&O 1997as) was prepared to provide a standardized approach to development of remote operating features and equipment for the repository surface facilities.

The following activities will be performed in FY 1998:

- Prepare a technical document that documents the integrated material handling simulation model for surface and subsurface waste handling operations using the WITNESS simulation program

- Prepare analyses that describe the support systems within the waste handling building, waste treatment building, and carrier preparation building (e.g., *Water Systems/P&ID Analysis, Gas Supply Systems/P&ID Analysis, Air Systems/P&ID Analysis, Power Supply/1-Line Analysis*)

- Provide space programming and resulting general arrangement layouts

- Provide initial assessment of shielding requirements, dose assessment, and as low as reasonably achievable requirements.

4.2.2.4 Site

The *Repository Surface Design Site Layout Analysis* (CRWMS M&O 1997at) established the repository arrangement of the surface facilities and features near the north portal. The analysis updated and expanded the north portal area site layout presented in the *Advanced Conceptual Design* (CRWMS M&O 1996b) to address resizing of the waste handling building, waste treatment building, carrier preparation building, and site parking areas; addition of the carrier washdown building; elimination of the cask maintenance facility; and development of a concept for site grading and flood control. The analysis also established the layout of the surface features (e.g., roads and utilities) that will connect all repository surface areas and located an area for a potential lag storage facility. Attachment 1 of the analysis includes the layouts as repository surface design site layout figures.

The *Repository Surface Design Site Layout Analysis* (CRWMS M&O 1997at) will be updated in FY 1998 to reflect changes to the requirements and design development for all repository surface areas.

4.2.2.5 Support Systems and Facilities

Support Systems and Facilities include the site systems and associated facilities. The systems include utility systems, safety and security systems, management and administration systems, the general site transportation system, and the site-generated hazardous and non-hazardous waste disposal system. Site support facilities include the transporter maintenance building, the administration building, security stations, the medical center, the computer center, the central warehouse, central shops, the motor pool, the mock-up building, the utility building, and the visitors' facility.

Although no analyses were performed on these systems and facilities during this reporting period, repository engineering files include preliminary data that will form the basis of a future report. The data will provide estimates of bounds of the required utility systems for the site (see Section 4.1.6).

The following work will be performed in FY 1998:

- Perform a staffing analysis

- Develop architectural/structural descriptions, operations descriptions, adjacency analysis, and general arrangements.

4.3 ENGINEERED BARRIER SEGMENT/WASTE PACKAGE DESIGN

This reporting period included a significant milestone for waste package development with completion of the EBS/waste package viability assessment design. Several deliverables and products were completed to address critical issues in order to achieve this goal. Completion of the viability assessment design was documented by the *EBS/WP Phase I Design Completion Letter* (CRWMS M&O 1997au).

Another important accomplishment was issuance of Revision 1 of the *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997b). The report describes the analysis methodology planned for use in demonstrating postclosure criticality control for the repository. The report captures the current status of the analysis methodology assumed in the waste package viability assessment design, and it supports both the viability assessment and the license application.

Most of the analyses conducted this reporting period were associated with validation of the viability assessment designs and evaluation of the cost of those designs. The results from some of the studies will be used in the TSPA-VA.

4.3.1 Waste Package Design

Waste Package Design Configurations – The study, *Determination of Waste Package Design Configurations* (CRWMS M&O 1997av), addressed the determination of the most cost-effective method for disposal of commercial spent nuclear fuel based on thermal and criticality goals. The study also addressed the 10 percent of the waste stream that was not addressed in previous studies. (This 10 percent represents those assemblies that exceed the criteria for the design basis fuel. Examples include the young fuel that exceeds thermal heat output or criticality goals, failed fuel, and long fuel assemblies.) The waste package design configurations necessary to dispose of 100 percent of the waste are:

- 21 pressurized water reactor assemblies with no absorber
- 21 pressurized water reactor assemblies with absorber plates
- 21 pressurized water reactor assemblies with absorber rods
- 12 pressurized water reactor assemblies with no absorber
- 12 long pressurized water reactor assemblies with absorber plates
- 44 boiling water reactor assemblies with no absorber
- 44 boiling water reactor assemblies with absorber plates
- 24 boiling water reactor assemblies with thick absorber plates.

Waste Package Size Study Support – Thermal, shielding, and criticality analyses were documented in the *Analytical Support for WP Size System Study* (CRWMS M&O 1997aw). These analyses concluded that shielding the waste package for radiation worker protection is not feasible from either a technical or a cost perspective. Thermal analyses showed little difference in waste package performance between large and small packages. Criticality analyses demonstrated little difference in criticality potential between small and large packages. Burnup credit and neutron absorbers are required for nearly all waste package sizes. However, from an operational perspective, there is less radiation exposure in handling large waste packages, because of fewer numbers of packages. From a cost perspective, the larger waste packages are less expensive. Therefore, the study verified the choice of predominantly large waste packages and did not recommend future designs using exclusively small containers.

Thermal Expansion Analysis – An evaluation of the thermal expansion of components of the 21 pressurized water reactor waste package configuration determined the required gaps between the internal components. The results of this evaluation provided the sizing of the basket plates. The evaluation is documented in the *UCF WP Static Loads, Thermal Expansion, and Internal Pressure Analysis* (CRWMS M&O 1997ax), which contains a schematic for the sizing of the basket plates.

Commercial Spent Nuclear Fuel Data – A literature search was completed to compile the dimensions and weights of commercial spent fuel assemblies. The data confirmed that the disposal container cavity lengths and basket cell widths are satisfactory, for all uncanistered commercial fuel waste packages in the viability assessment design. The fuel assembly weights (masses) will be used in structural and handling evaluations of the waste packages. *The Waste Container Cavity Size Determination* (CRWMS M&O 1997ay) contains a detailed description of this work.

Thermal Design Basis Fuel – Each waste package design is part of a family of designs that will dispose of 100 percent of the commercial spent nuclear fuel waste stream. A specific design basis fuel was specified for each waste package design. An evaluation, *Preliminary Design Basis for WP Thermal Analysis* (CRWMS M&O 1997az), determined the design basis fuel for the thermal design. The definition of the design basis fuel and all the associated caveats are extensive. Generally, the primary waste package is large, the secondary waste package is small to cover the hottest fuel, a waste package with control rods was selected for the most reactive fuel, and a long waste package was required for the South Texas SNF. The results of this evaluation will be used for the fuel assembly heat output for future thermal analyses.

Additional Barriers Study – A study was completed on the effects and benefits of applying additional barriers to the waste package. Additional barrier studies in the engineered barrier system are based on the *Controlled Design Assumptions Document* (CRWMS M&O 1997d) TDSS 025 and TDSS 026, which state that the environment in the repository may have a higher potential for percolating water to seep into the emplacement drifts than previously stated. Additional barrier options were considered, including an integral drip shield, a separate drip shield, and backfill. Also, filter materials in the invert media were considered as a means to collect corrosion products and fission products. Structural, thermal, and thermal stress evaluations were performed. While each design had some problems, the integral drip shield showed the most promise, especially that using alloy C-22 or coating layers that are predominately metallic, ceramic, or an intermediate combination.

The additional barriers will continue to be evaluated as part of the total system performance assessment to better quantify their potential benefits. The drip shield and backfill options noted above are identified among the design options that will be pursued as part of the viability assessment, in comparable detail with the reference design. Sensitivity cases on each option will be performed to determine the merit of each option, and to determine which should be further pursued in the waste package design. References include *Rock Fall Analysis of an Additional Barrier on UCF WP* (CRWMS M&O 1997ba), *Thermal Evaluation of the 21 PWR UCF Waste Package with an Additional Barrier* (CRWMS M&O 1997bb), and *Additional Barriers Letter Report* (Benton letter).

4.3.2 Waste Package Fabrication

The waste package engineering development program for FY 1997 was completed. The program proved, using a mock-up, that it is viable to use the shrink-fit method to manufacture the waste package barriers. The mock-up was sent to Lawrence Livermore National Laboratory, where tests will be conducted to determine the contact between the cylinders, galvanic properties, and corrosion resistance (see Section 4.3.5.1).

The program consisted of a closure weld methods program and a nondestructive examination methods program. The mock-up duplicated the top part of the waste package. The materials used were A516 Carbon Steel and Alloy 625. Fabrication of the cylinders was accomplished by the shrink-fit method, and the interface of the two barriers was examined using ultrasonic, dimensional measurements, and visual inspection. Both the inner and the outer lids were welded using the hot wire automatic tungsten inert gas process, which decreased the welding times for the closure weld because of higher deposition rates than those assumed for the cold wire automatic tungsten inert gas process that was used in the FY 1996 program. The results of these programs are published in the *Waste Package Closure Methods Report* (CRWMS M&O 1997bd) and the *Nondestructive Examination Methods Report* (CRWMS M&O 1997be).

Revision 2 of the *Waste Package Fabrication Process Report* (CRWMS M&O 1997bf) was completed. This identified shrink fit as the preferred method of fabrication, as a result of the development program. Cost estimates for four of the waste package designs were obtained from two different fabricators. These estimates verified the Waste Package Engineering Development estimates.

4.3.3 Drawings

Drawings of preliminary viability assessment designs were released for the following waste package configurations:

- 21 pressurized water reactor with absorber plate
- 44 boiling water reactor with absorber plate
- 12 pressurized water reactor, no absorber
- 4 defense high-level waste
- 5 high-level waste.

Each drawing package contains approximately 20 drawings. These drawings encompass the waste package design for the baseline commercial spent nuclear fuel and approximately 80 percent of the anticipated waste stream. The drawings are available in the Project's Records Information System.

4.3.4 Waste Package Materials & Waste Forms

4.3.4.1 Non-Commercial Waste Forms

In 1996, the baseline was expanded to include spent nuclear fuel from defense production reactors, naval reactors, and domestic and foreign research reactors. Since mid-1996, a significant effort has concentrated on integrating these forms of spent nuclear fuel into the design and operation of the MGDS. Periodic meetings have been held with representatives of the Assistant Secretary for Environmental Management and with representatives of the U.S. Navy to facilitate the planning and integration of these additional types of spent nuclear fuel into the CRWMS for disposal. A multi-year program has been planned to develop conceptual designs for waste packages to accommodate these additional types of spent nuclear fuel.

The first type of spent nuclear fuel selected for analysis is the aluminum-based spent nuclear fuel from research reactors. The conceptual designs were developed for spent nuclear fuel canisters and waste packages for the disposal of aluminum-based spent nuclear fuel with uranium enrichments of 20 and 93.5 percent U-235. These designs are included in the viability assessment design. A report was issued summarizing the criticality, shielding, thermal, and structural analyses that support the conceptual designs of the canisters and waste packages for the intact aluminum-based spent nuclear fuel, *Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel; Phase I, Intact Codisposal Canister* (CRWMS M&O 1997bg).

4.3.4.2 Waste Acceptance Criteria

Acceptance criteria for commercial fuel and defense high-level waste glass were developed in the *Mined Geologic Disposal System Disposability Interface Specification* (CRWMS M&O in prep [e]). This document provides a centralized list of waste properties and documentation that must be met in order for wastes to be accepted for disposal in the MGDS. The information will also be used to support the design, as well as the TSPA-VA.

4.3.4.3 Waste Package Supports and Sorptive Inverts

Evaluations were performed to determine the material to be used for the waste package supports and to evaluate the addition of sorptive materials to the inverts. The analysis *Waste Package Supports and Sorptive Inverts - Materials Selection* (CRWMS M&O 1997bi) documented this work. Sorptive invert materials were evaluated in this analysis because the addition could contribute to isolation of waste from the accessible environment.

Carbon steel was selected as the best material for waste package supports. Two sorptive minerals with potential as sorptive additives to the supports, apatite and envirostone, were evaluated qualitatively against selection criteria for sorption effectiveness and cost. Pending results of further studies, no sorptive additives were recommended.

This work supports the TSPA-VA and both the viability assessment and the license application designs.

4.3.5 Waste Package and Waste Form Testing and Modeling

4.3.5.1 Waste Package Degradation Model Abstraction and Testing

Following the waste package degradation model abstraction workshop conducted in January 1997, work in model development focused on providing a "skeleton" of degradation models that will be operative over the long containment and controlled release periods. Scenarios for waste package container degradation were developed by the workshop participants, and a sequence of degradation events was proposed. This sequence has been the basis for the model abstractions that will form the "backbone" for the TSPA-VA. Of particular note is the approach being taken at the outer barrier/inner barrier interface, where a number of degradation modes occur in juxtaposition. Residual stress from container fabrication and welding plus the aggressive environment can lead to cracking of susceptible materials and microstructures. The composite model was advanced to show how the various modes interact and to calculate limits on pH and other chemical concentrations in the interface region.

During this reporting period, several deliverables were completed as input to the performance assessment. These included summaries of recent test data and performance predictions for inner barrier performance, including pitting corrosion, crevice corrosion, stress corrosion, and galvanic corrosion. A very comprehensive report on the corrosion effects at the outer barrier/inner barrier interface was completed and is in review (Huang in prep.).

OCRWM established an expert elicitation on waste package degradation that brought corrosion experts, both internal and external, to the project to obtain an additional set of recommendations for the total system performance assessment representation of container degradation. The panel was formed to provide estimates of various degradation modes and rates based on generally known phenomenology about the waste package candidate container materials. The panel was also asked to provide its assessment of the uncertainty of its estimates and to describe the variation that would occur in the degradation rates between different types of containers and at various locations on an individual container. The panel members provided individual elicitations with guidance from the Project organization team. The issues asked of the waste package degradation expert elicitation panel were cross-linked with, and developed from, the model abstraction workshops. A final report is in preparation.

The long-term comprehensive corrosion test, begun in September 1996, continued through this reporting period, and 12 additional vessels were brought on line in the past six months. Currently, 24 vessels are in full operation. Of the four test environments used, three consist of variations in chemical concentrations and pH. The fourth, developed in the last reporting period using input from investigators evaluating the performance of concrete materials in the repository, simulates the effect of leaching through a concrete liner. The work so far provides preliminary indications that the concrete will react with water and carbon dioxide, resulting in carbonation of the concrete. By the time the repository has cooled enough to permit aqueous intrusion into the drift, the concrete is expected to be mostly converted to carbonate phases. Water contacting this aged material will not leach out free alkali but will become more like a limestone-saturated water. This conclusion is based on experimental work and on calculations made from the geochemical modeling code.

Single-metal tests continue and are comprised of three geometries: stressed U-bends for stress corrosion and hydrogen embrittlement evaluation, intentionally creviced specimens for crevice corrosion determination, and weight loss specimens for general corrosion measurement. The last two configurations also offer surfaces on which pitting corrosion, intergranular corrosion, and selective leaching can be observed and measured in susceptible materials.

Galvanically-coupled specimens were designed and were recently emplaced in the new vessels. The carbon steel specimens extend out beyond the corrosion resistant material. This arrangement allows study of the "throwing power" effect between components, as well as the crevice effects between the two components.

With all 24 vessels operating, some 16,000 specimens are in various stages of testing. Specimens from several of the vessels have already reached the six-month exposure interval, and two of the vessels have just reached the one-year interval. These specimens are in the process of being characterized to measure the amount of general corrosion, and examined for evidence of other forms of corrosion. Some initial characterization work has been performed on the carbon and alloy steel specimens exposed to the dilute and the concentrated water. The general results indicate that more severe corrosion occurs in the "dilute" (low ionic strength - approximating 10x the solute concentration in J-13 well water) water environment. The corrosion rate was 93 μm/yr for A516 Grade 55 carbon steel at 60°C, and 100 μm/yr at 90°C after six months' exposure in the dilute water environment. In comparison, the corresponding rates were 66 μm/yr and 12 μ m/yr in the "concentrated" (high ionic strength, approximating 1000x J-13) water for the same two test temperatures. The lower corrosion rate in the more "concentrated" water is likely due to the formation of protective carbonate-rich scales and decreased solubility of oxygen. Both oxygen solubility and calcium carbonate solubility decrease with increasing temperature. Further, even for the "dilute" water environment, the two corrosion rates are rather similar at the two temperatures. This is probably because of the offsetting effect of decreasing oxygen solubility, and possibly due to the decreasing calcium carbonate solubility against generally increasing effects of the metal dissolution kinetics. Preliminary indications from the one-year exposures indicate that the corrosion rates are decreasing with exposure time. The results to date follow generally observed behavior of the aqueous corrosion of carbon steel. Some of these results were reported in milestone WP60107 (McCright 1997b).

Additional analysis/testing continues in the following activities:

- Thermogravimetric analysis work continues with the intent of determining the temperature/humidity/surface conditions where transition from dry oxidation to humid air corrosion takes place.

- Electrochemically based testing continues, particularly for determination of the localized corrosion resistance of candidate materials for the inner barrier. Work is also progressing on maintaining potentiostatic conditions on the metal specimens to determine pit growth kinetics. Results from these experiments were reported in *Material Thermal Stability* (McCright 1997a) and *Status of Self-Loaded Crack Growth Tests* (Roy 1997a). Electrochemical methods are being used to measure the galvanic potentials and currents between carbon steel specimens and specimens of the candidate inner barrier materials in different environments, in different temperatures, and in different area ratios between the two metals. These results were reported in *Status of Electrochemical Galvanic Corrosion Testing* (Roy 1997b).

- The effect of microbiologically influenced corrosion is being determined, principally on carbon steels, Alloy 400, and 70/30 Cu/Ni materials. A new test cell, which accommodates a much larger test specimen, has been developed so that the morphology of the microbiologically influenced corrosion attack can be measured. Results from the microbiologically influenced corrosion studies were provided in McCright (1997a).

- Work began on evaluating the microstructures of the waste package container candidate materials and changes that could occur in these microstructures over the long containment and controlled release period. Of particular concern are the microstructures in the weld and heat-affected zones. Results of the investigations into the shrink-fit mock-up fabrication (see Section 3.3.2) were reported (McCright 1997a).

- Fracture-mechanics type stress corrosion cracking specimens are being exposed to aggressive test solutions. Results indicate that Alloys 825 and 625 are susceptible to stress corrosion cracking in acid chlorides at 90°C. Alloy C-22, tested under comparable conditions, has not shown any crack propagation thus far. The results are summarized in *Status of Self-Loaded Crack Growth Tests* (Roy 1997a).

- Thermally-sprayed ceramic materials used as coatings on one of the metal barriers represent a design/materials option (as discussed in Section 4.3.1) for dealing with the possibility of higher water fluxes into the repository. Mechanical testing of steel blocks coated with alumina was performed using a drop tower, in which a penetrator made of natural or simulated tuff was impacted onto the test specimen and the energy of the impact measured. Results indicated that only natural tuff impactors with an irregular geometry could damage the ceramic coating. These results were reported in *Information on Ceramic Material Tests for Viability Assessment and Design* (Wilfinger 1997). More recently, ceramic-coated specimens have been placed in one of the vessels of the long-term corrosion test facility (90°C, dilute water environment).

- Work continues in the evaluation of criticality control materials that will be used for the basket and other internal components in spent nuclear fuel waste packages.

The major activity for FY 1998 is continuation of the experimental work to provide input for waste package disposal container model development. Many of the tests have matured to the level at which abundant data are forthcoming. The corrosion performance models have also reached an advanced level at which the test data will have immediate application.

4.3.5.2 Waste Form Testing and Modeling

The waste form testing and modeling technical activities obtain degradation, dissolution, alteration, and release rate data from specific experiments performed on commercial spent nuclear fuels and defense high-level waste glasses. For these waste forms, rate response models are developed that use the physical processes observed in testing as a conceptual basis, and use the test data to evaluate material parameters and coefficients in the function forms of models. The testing data, model development, and model predictions were incorporated in the revisions of the *Waste Form Characteristics Report* (Stout and Leider 1997). The information is used by performance assessment for incorporation into its models for the total system performance assessment.

Oxidation of Spent Nuclear Fuel – Spent nuclear fuel oxidation test data showed a U4O9 to U3O8 oxidation rate dependence that decreased with increasing spent nuclear fuel burnup for fuels in the middle range of burnup. The rate response at the high burnup (~60 MWd/kgU) of the spent nuclear fuel inventory remains to be tested to see if this trend continues. The decreasing oxidation rates with burnup mean that the existing model in the *Waste Form Characteristics Report* (Stout and Leider 1997) would be conservative with respect to the spent nuclear fuel burnup attribute variable. Spent nuclear fuel oxidation is a degradation or alteration mode that can significantly increase the potential radionuclide release rate in the repository, due to splits in defected zircalloy cladding resulting from volume increases during the oxidation phase change. The rate of oxidation from U4O9 to U3O8 is very slow for temperatures below 100°C. This rate is much less than a corresponding dissolution rate if the fuel were wetted, and at these temperatures atmospheric oxidation is not expected to be a problem.

Spent Fuel Dissolution, Alteration, and Release Testing – Testing is being conducted to provide data for modeling the effects of burnup and radiolysis chemistry on dissolution rates. Preliminary results suggest that the high surface area-to-water volume batch tests reproduce the oxy-hydroxide and uranyl silicate alteration phase formation observed in unsaturated drip tests with oxygenated groundwater. These alteration phases have a significant increase in volume relative to that of the spent nuclear fuel and may also result in additional opening/splitting of defected cladding. The fuels in the tests have undergone five years' reaction time at 90°C as of the end of September 1997. The vapor tests are showing a mode of aqueous release due to vapor transport and condensation on the spent nuclear fuel surface. The measured mass of actinides aqueously released during the unsaturated tests are being analyzed to determine wetted film concentrations on the surface of the spent fuel. These are limiting concentrations for release rates at low-flow water rates expected in the repository. Concentrations are similar to and sometimes orders of magnitude less than the solubility limits that were previously used. To augment the existing unsaturated drip tests, a batch test method with high fuel surface area-to-water volume ratios (surface area-to-water volume at 5000 m^{-1}) is being developed to measure the solution composition in contact with spent nuclear fuel during the reaction process. Information from these tests is needed as input for model development and confirmation.

An updated intrinsic dissolution model based on nonequilibrium thermodynamics was provided for the total system performance assessment and will be incorporated in the next revision of the *Waste Form Characteristics Report* (Stout and Leider 1997). Unsaturated test data and alteration product identification were also provided for the total system performance assessment, as stipulated in the TSPA-VA abstraction plans.

Models for Release from Spent Nuclear Fuel – Rate response models are being developed for the oxidation, degradation, dissolution, and radionuclide release rates of spent nuclear fuel waste forms. The existing model was modified slightly, by including linear terms for all five independent variables. The model is being extended to address more explicitly radiolysis effects on water chemistry and is being reformulated to account for the heterogeneities of different surface dissolution reactions.

Model development to describe release rates based on the unsaturated test data continued. The unsaturated test data indicate that a significant alteration layer has evolved on the fuel surface. Using the unsaturated vapor test, as well as the low and high drip-rate test data, preliminary values for film concentrations and flow-dependent unsaturated release rates are being determined.

Thermogravimetric analysis oxidation tests, flow-through dissolution tests, and unsaturated alteration/release tests on spent nuclear fuel will continue in FY 1998. The long-term unsaturated tests will continue with the addition of high surface-to-volume batch tests to strengthen coupling of the unsaturated tests with model development. Other tests will be initiated to characterize colloidal species and to assess the effects of potential volume increases resulting from the alteration of UO2 and the formation of additional mineral phases in open and split cladding. Samples from high-burnup spent nuclear fuel will be added to ongoing testing matrices to evaluate potential burnup dependence. The development of submodels for release rate terms in the mass transport model will continue. Submodels of radiolysis and surface alteration effects will be refined. They will be consistent with the increasing sets of dissolution/release data. The spent nuclear fuel data and models will be incorporated in revisions to the *Waste Form Characteristics Report* (Stout and Leider 1997).

Glass Waste Form Testing and Modeling – Glass waste form testing activities provide data on radionuclide release mechanisms and degradation/alteration rates of glass waste forms. These data are being used to constrain and guide ongoing model development for glass corrosion, for use in the total system performance assessment.

Long-term unsaturated tests (drip tests) of two glass compositions (Defense Waste Processing Facility and West Valley ATM-10) continue in two test series labeled the N2 (11.5 years duration) and N3 series (10 years duration). Analyses of N2 and N3 test results include detailed characterization of soluble and colloidal release of actinides and fission products. Flow-through tests on SRL-202 glass compositions in ferric iron-containing solutions are in progress. This test series will provide data to support YMP waste form materials selection activities. The chemical modeling of glass degradation is being used to synthesize results from ongoing experimental work, determine the rate-limiting chemical mechanisms controlling glass alteration rates, and provide a mechanistically-based method for making long-term predictions of glass degradation to be incorporated into the total system performance assessment models.

Unsaturated testing and flow-through tests of defense high-level waste glasses will continue in FY 1998. The data from these tests will be used to guide and evaluate glass response model development. The data and models will be incorporated in revisions to the *Waste Form Characteristics Report* (Stout and Leider 1997).

4.3.5.3 Engineered Barrier System/Near-Field Environment Performance Assessment Abstraction/Testing Interface

During this reporting period, engineered barrier system/near-field environment performance assessment worked in collaboration with the process expert staff from the waste package and site investigation activities on the development and abstraction of models for use in the TSPA-VA. Related to near-field geochemistry and water flow through the engineered barrier system, new models are being developed to represent processes that were not included in previous total system performance assessments. Chapter 3, Section 3.2.5 presents details. The abstracted models for use in the TSPA-VA base case are scheduled for implementation by November 15, 1997. As part of the documentation of this process, a deliverable report was completed: *Waste Form Degradation and Radionuclide Mobilization Workshop Results* (CRWMS M&O 1997bj).

FY 1998 activities will include completion of the model abstractions, assisting in the implementation of these models in the TSPA-VA, assisting in sensitivity analyses for the TSPA-VA, and beginning documentation for the TSPA-VA report.

4.3.6 Criticality

4.3.6.1 Disposal Criticality Analysis Methodology Development

Substantial progress occurred during this reporting period in developing the methodology for performing disposal criticality analyses. With the methodology becoming more complete, a preliminary sample application can be run on a Waste Package design during the next reporting period. This will help determine the adequacy of the criticality control(s) and if the methodology works as anticipated.

A broad range of features, events, and processes important to criticality is used to generate preliminary scenarios. Configurations from the preliminary scenarios are evaluated using neutronics models. Specific configurations and probability distributions from these scenarios are used to determine the range of potential critical configurations. The risk associated with repository criticality is the product of the probability of occurrence and the consequences, summed over possible events. Evaluations are performed for three locations: within the waste package, in the near field but outside the waste package, and in the far field.

Validation of the models uses data from chemical assays of irradiated fuel and from measurements in laboratory assemblies and commercial nuclear power plants. Development and validation of the models used in the methodology is a continuing process that has not been completed.

An informal meeting with NRC staff (an Appendix 7 meeting) was conducted to discuss issues and disposal criticality methodology, and to exchange information. The neutronics model, initially developed in 1995, was updated and is near its final form. A revision of the *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997b) was issued. The revision was a major change to the original report, because of the evolution of how models are implemented in the methodology. This work moves the project closer to submittal of a disposal criticality analysis methodology topical report, scheduled for FY 1998, that will present the methodology for performing disposal criticality analysis (including the use of burnup credit) for NRC acceptance. Upon NRC acceptance, the DOE will use the methodology in the license application to demonstrate acceptability of the proposed criticality control system.

Commercial reactor critical data, laboratory critical experiments, chemical assay data, and degraded waste package systems were the focus for development and verification of the neutronics model for this reporting period. The four completed reports that support the *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997b) include:

- *Summary Report of Commercial Reactor Critical Analyses Performed for the Disposal Criticality Analysis Methodology* (CRWMS M&O 1997bk). This report contains the summary of analysis results for spent fuel benchmark criticals. The spent fuel benchmark criticals are an important part of the validation of burnup credit for the methodology.

- *Summary Report of Laboratory Critical Experiment Analyses Performed for the Disposal Criticality Analysis Methodology* (CRWMS M&O 1997bl). This report contains the summary of analysis results for standard benchmark criticals. Results from these analyses, and the analyses for the spent fuel criticals, will be used to determine the upper subcritical limits for various postclosure configurations.

- *Summary Report of SNF Isotopic Comparisons for the Disposal Criticality Analysis Methodology* (CRWMS M&O 1997bm). This report contains the summary results of comparisons between calculated and measured isotopic concentrations for samples of spent fuel. The results are used in the validation of the models for burnup credit.

- *Degraded Waste Package Criticality: Summary Report of Evaluations through 1996* (CRWMS M&O 1997bn). This report summarizes the evaluations used in developing the probabilistic parts of the disposal criticality methodology.

Additional reports and analyses completed this reporting period provided supporting information. Some of the more significant include *Summary Report of Commercial Reactor Criticality Data for Crystal River Unit 3* (CRWMS M&O 1997bo), *Summary Report of Commercial Reactor Criticality Data for McGuire Unit 1* (CRWMS M&O 1997bp), and *Summary Report of Commercial Reactor Criticality Data for Sequoyah Unit 2* (CRWMS M&O 1997bq).

4.3.6.2 Chemical Assay Work

Chemical assay work was one aspect of the neutronics model (see Section 4.3.6.1). The results of this work will be used to support development and validation of the isotopic models for commercial spent nuclear fuel. These models and their validation will be presented in the disposal criticality analysis methodology topical report in FY 1998.

The *Chemical Assay Data Letter Report* (CRWMS M&O 1997br) summarized the chemical assay analyses performed this year in support of the *Disposal Criticality Analysis Methodology Technical Report* (CRWMS M&O 1997b), and the available chemical assay data sources identified by Oak Ridge National Laboratory for the Project (*Evaluation of Measured LWR Spent Fuel Composition Data for Use in Code Validation*, 9/5/97, draft, ORNL).

4.3.6.3 Analysis of Degraded Waste Packages

Results of studies conducted relating to the criticality potential that arises from degradation of waste packages were reported in FY 1997 (CRWMS M&O 1997bs; 1997bt; 1997bu; 1997bv; 1997bw). The results of these studies were summarized in the *Waste Package Probabilistic Criticality Analysis: Summary Report of Evaluations in 1997* (CRWMS M&O 1997bx). Benefits from these studies include:

- Narrowed the uncertainty in estimation of probability of criticality for commercial spent nuclear fuel

- Furthered development of the degraded mode criticality methodology for the disposal criticality analysis methodology topical report (planned to be prepared)

- Narrowed the uncertainty in estimation of the direct consequences of a criticality (should such a criticality occur), which is input to the TSPA-VA

- Demonstrated that there are no likely mechanisms for accumulating a critical mass in the invert beneath a waste package

- Showed that internal criticality can be avoided for the immobilized plutonium waste form by limiting the plutonium loading in the waste packages.

4.4 ESF

The ESF is being designed and constructed to allow performance of a program of in situ exploration and testing above, at, and below the depths at which waste might be emplaced. This work will make a major contribution to determining the suitability of the Yucca Mountain site for construction of a potential underground high-level nuclear waste repository. This work also addresses the 10 CFR 60.15(b) requirement for in situ exploration and testing at the depths that waste would be emplaced.

The ESF work can be divided into two parts. The first part includes design of the equipment and systems necessary to perform the ESF functions. Basically, this is the design to support excavation activities, including the design of utilities and support equipment. The second part is construction and excavation that will implement the design. Chapter 5 describes testing and analysis activities conducted in the ESF.

4.4.1 ESF Design

The Nuclear Waste Technical Review Board has consistently expressed a position that high-priority site characterization activities include a full east-west traverse of the proposed repository block. In agreement with this position, the Project has undertaken planning and scheduling activities to implement a program for enhanced characterization of the repository block through incorporation of a new drift in the ESF (see Adams letter). An Integrated Planning Committee was formed to examine the concept, identify potential project objectives, and define the technical basis for enhanced characterization of the repository block activities. The results of the Integrated Planning Committee are being compiled in the *Final Report on Enhanced Characterization of the Repository Block Planning Effort* (CRWMS M&O in prep. [g]).

The enhanced characterization of the repository block cross drift is expected to provide data enhancement in three general categories: scientific data used for the total system performance assessment/license application, scientific or construction data used for the license application design, and repository construction planning. The following key points have been identified:

- Hydrologic testing and sampling for environmental isotopes and fracture-filling minerals below an identified zone of high surface infiltration are important. This testing and sampling will support investigation of possible correlation of high surface infiltration and fast flow paths at depth.

- Displacement on the Solitario Canyon Fault increases dramatically from north to south. Characterization of the Solitario Canyon Fault is needed at the fault location where displacement is great enough to see well-developed physical characteristics of the fault zone and wall-rock deformation associated with the fault and to evaluate the hydrology. The location and characterization will be used to determine the required stand off distance from this major fault for the repository design. Hydrology will be evaluated as discussed in the first key point.

- Very little data exists on the physical properties of the rock in the lower two subunits (lower non-lithophysal and lower lithophysal) at the repository level. The ESF main drift is located almost entirely within the middle non-lithophysal subunit. The repository will use the middle non-lithophysal as well as the lower lithophysal and the lower non-lithophysal. These latter two subunits lie below the middle non-lithophysal. The cross drift will traverse all three subunits providing significant hydrologic and geologic information.

- Fracture distributions and abundance vary both from north to south within the section, and between stratigraphic subunits, which makes it important to sample the entire section of the emplacement horizon.

- Testing in the Calico Hills formation would improve the understanding of flow and transport processes below the repository horizon.

- The splay extending from the Solitario Canyon Fault in the central part of the block shows decreasing displacement upsection in the outcrop. This possibility should be checked by underground construction.

Design of the launch chamber for the cross drift was started in this reporting period.

Designs for the ESF flow-through ventilation system, test niches 1 and 2, and the heated drift bulkhead for the Thermal Testing Facility were completed. Also, the design confirmation program resulted in issuance of the *Controlled Drilling and Blasting Design Confirmation Analysis* (CRWMS M&O 1997by), and the *Key ESF Design/Geomechanics Parameters Confirmation* (CRWMS M&O 1997bz). Support for installation and checkout of the data collection system for the Drift Scale Test continued during this reporting period. Two key support efforts were equipment expediting and software development for the Data Collection System. These efforts were crucial in supporting the milestone for heater turn-on.

The following tasks are scheduled for FY 1998:

- Complete the ground support confirmation program by issuing the following documents: Thermomechanical Classification, Probabilistic Key Block Analysis, Ground Support Reconciliation, Key Design Parameter Confirmation Analysis, and the Steel Set Load Confirmation Analysis

- Complete the launch chamber and drift design of the cross drift

- Complete the design of the three test alcoves and two test niches for the cross drift

- Complete the design of test niches 3 and 4 in the ESF Topopah Spring main drift.

4.4.2 ESF Construction

Construction of the south portal box cut for the ESF Topopah Spring tunnel loop was completed during the first week of this reporting period. On April 25, 1997, the tunnel boring machine broke through to the surface at the south portal, completing excavation of the 7,877-m (25,844-ft or 4.9-mi) ESF loop.

Other important milestones reached during this period were completing excavation of the Northern Ghost Dance Fault Alcove and drift scale flow test niches 1 and 2. Completion of these excavations allowed testing to commence in these alcoves and niches. Chapter 5 discusses testing in alcoves and niches. Construction support for the Drift Scale Test in the Thermal Testing Facility continued with placement of concrete lining, drilling test holes, and installation of testing equipment in the facility. Excavation of the Southern Ghost Dance Fault Alcove was almost complete at the end of this reporting period.

The potential impacts of construction related factors on the design of the enhanced characterization of the repository block cross drift were evaluated and presented in constructability reviews during the design process. Specific items addressed included tunnel boring machine design criteria, design of the tunnel boring machine launch chamber, and geometry of the proposed cross drift.

The following tasks are scheduled for FY 1998:

- Complete excavation of the Southern Ghost Dance Fault Alcove
- Complete construction support for the Drift Scale Test in the Thermal Testing Facility
- Complete excavation of test niches 3 and 4 in the ESF Topopah Spring main drift
- Complete excavation of the launch chamber and the cross drift
- Complete construction of the Busted Butte facilities consisting of an access road, highwall, and excavation of approximately 75 m of drifts.

CHAPTER 5 - SITE CHARACTERIZATION

Site characterization includes hydrological, geological, geochemical, and geo-engineering studies of Yucca Mountain to describe the natural features, events, and processes of the site (Figure 5-1). The information is needed to design and evaluate the expected performance of the potential mined geologic disposal system (MGDS). One of the principal uses of the descriptions is to develop models for the performance of engineered barrier systems and models for the nature and rate of radionuclide movement after the eventual release of radionuclides from the disposal system.

Key Advances

Synthesis and modeling activities have produced significant results that support the viability assessment and represent important advances in the development of quality-assured data and models to support the license application. Key advances during the period include the following:

Altered Zone and Near-Field Environment – Preparations for the heating phase of the Drift Scale Test are on schedule. Heating should begin in early December 1997. Results of recent field and modeling studies indicate that expected changes in altered zone mineralogy and flow paths, which are expected to result from coupling of heat from emplaced waste and fluids in the altered zone, could increase sorption of radionuclides and increase groundwater transport times near the potential repository and through the underlying part of the unsaturated zone. Such increases have the potential to improve system performance. Waste package interaction with the near-field environment is sensitive to the quantity, temperature, and chemical content of both liquid water and water vapor as well as drift-gas composition. Two studies indicate that little or no fracture flow is likely to occur into the drifts at the percolation fluxes that are believed to exist at Yucca Mountain. Alteration of inflow (fracture flow) is not seen as a major determinant of waste package life, because little or no water is expected to contact the waste packages. Relative humidity (controlled by porewater) appears to be a significant environmental variable affecting the corrosion rate of the outer barrier of the waste package.

Some preliminary results were obtained from the Large Block Test at Fran Ridge and the Single Heater Test in the Exploratory Studies Facility (ESF). For the Large Block Test, after heating was initiated, a drying zone developed around all the heaters, and moisture in the rock was significantly decreased. The heating element for the Single Heater Test was turned off in May 1997. Preliminary analyses of predicted and measured temperatures indicate that conduction is the dominant mode of heat transfer, although hydrologic processes have some effect. Water was mobilized by the heating, and the water left pore spaces in the rock, apparently as vapor, which then condensed in the fracture network. Initial results indicate that the condensate interacted with minerals in the fracture network prior to being sampled. Also, the capacity of the rock to transmit fluids appears to be related to the presence of fractures. Additional analysis of the data from these tests is in progress, and evaluations of the data for use by repository and waste package design and performance assessment have been initiated.

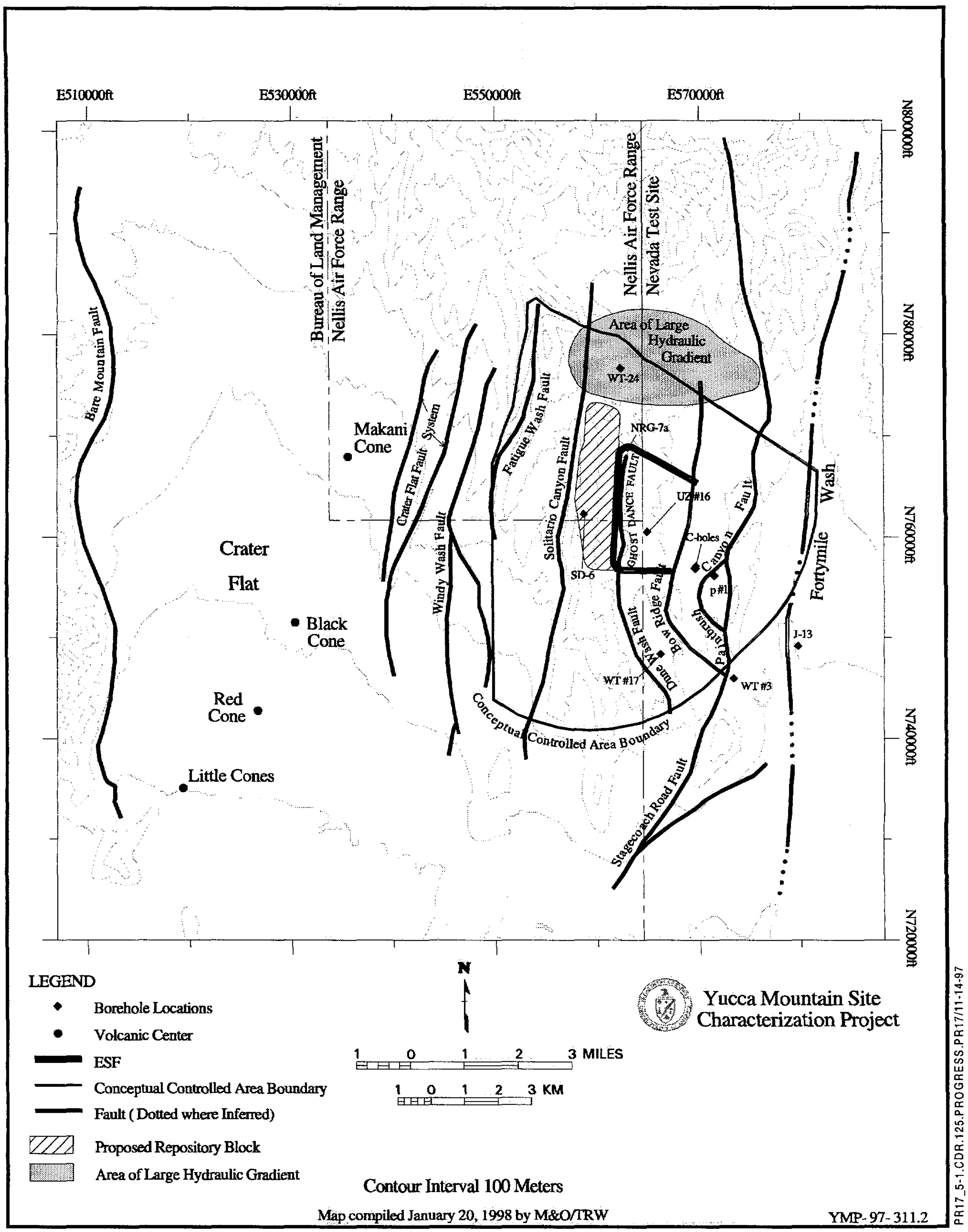

Figure 5-1. Map Showing Selected Features of the Yucca Mountain Site and the Surrounding Area

Site Unsaturated-Zone Modeling and Synthesis – Estimates of the maximum upper bound on present-day, spatially averaged percolation flux in the unsaturated zone were refined. Simulations showed that average percolation flux exceeding 20 mm/yr is inconsistent with available temperature, chloride content, and matrix property data. Chlorine-36 and tritium data indicated that fast pathways are associated with faults mapped at the surface and in the ESF. However, the fast paths appear to be restricted to the northern part of the ESF, because no samples with unambiguous levels of bomb-pulse chlorine-36 or tritium were collected in the southern part of the ESF. Permeability differences at the contact between the Tiva Canyon welded (TCw) and Paintbrush non-welded (PTn) hydrogeologic units corroborate indications about the potential for lateral diversion of water above the repository horizon. Preliminary results from moisture-balance calculations indicate that daily losses from evaporation are the same order of magnitude as the annual percolation rate. Therefore, potential exists for significant drying of the repository host rocks during construction and operation of a repository because of the presence of ventilation. However, the long-term effects of potential drying have not yet been determined.

Saturated-Zone Hydrologic System Modeling and Synthesis – An updated three-dimensional flow and transport model for the site-scale saturated-zone flow model was developed. The updated model more accurately represents the geometry of hydrogeologic units and is expected to provide a more accurate calculation of ground-water flow paths and fluxes. However, significant uncertainty remains in estimates of flux beneath the repository area, because the fluxes calculated by the site-scale model were greater than fluxes calculated by the regional flow model. The sensitivity of system performance to these uncertainties has not yet been determined.

Site Geochemical Synthesis and Modeling – Very small amounts (10^{-15} curies) of plutonium from a weapons test 29 years ago were detected, associated with natural colloids in groundwater at the Nevada Test Site (NTS). This was the first documented transport of plutonium in groundwater at the NTS, and the transport occurred in volcanic rocks similar to those at Yucca Mountain. Results of laboratory colloid studies indicate that at colloid concentrations typical of water from well UE-25 J#13, sorption of radionuclides onto natural colloids would have to be very high and irreversible before significant quantities of plutonium could be transported. Results of other laboratory studies indicate that sorption of radionuclides that form anions appears to be negligible in Yucca Mountain rocks that do not contain zeolites. If accurate, the result indicates that sorption in unsaturated zone rocks could be less effective in retarding some radionuclides. Other results indicate that radionuclide transport times increase as the degree of saturation in Yucca Mountain tuffs decreases. This result suggests that the performance of a drier repository would be improved relative to the performance of a repository developed in rocks with greater moisture contents.

Geologic Framework and Integrated Site Model – The geologic framework of the Integrated Site Model was updated using the Project's quality-assured, borehole geophysical database. This update contributed directly to the development of a qualified three-dimensional geologic framework model. Analysis of the ESF mapping for the southern part of the main drift and all the south ramp indicate that the stratigraphy and structure were very consistent with preconstruction predictions. These important results provide a basis for validating part of the Integrated Site Model and generally improve confidence in the accuracy of the model. The Integrated Site Model is being developed to provide the combined geologic, rock properties, and mineralogic framework to support hydrologic and transport models of Yucca Mountain.

Biosphere, Meteorology, and Paleoclimate and Paleoenvironmental Synthesis – Preliminary Biosphere Dose-Conversion Factors were calculated to support total system performance assessment-viability assessment (TSPA-VA). Studies assessed how radionuclides potentially released from the proposed repository could be transported through various pathways to humans. Sensitivity analyses were performed to identify the most important pathway parameters, and a regional survey was conducted to characterize the critical exposure group for the modeling effort. Analysis of the extensive data set documenting past climate within the Yucca Mountain region indicates that past climates changed from interglacial conditions to glacial conditions in less than 1,000 years. Changes within a glacial or interglacial period, from wetter to drier or colder to warmer conditions, occurred in less than 300 years. Interpretation of paleoclimate data also suggests that the long-term average levels of effective moisture during the past 400,000 years were higher than today about 80 percent of the time. Modern arid climates of the Holocene began about 8500 years ago and are the driest and perhaps warmest of all past interglacials. Future climate is expected to begin moving towards glacial conditions and away from present-day interglacial arid modes within the next thousand years. If these interpretations are accurate, infiltration, percolation flux, water table elevations, and groundwater flow during the postclosure performance period could be greater than those predicted based on the present-day climate.

Disruptive Conditions – Work continued on the probabilistic seismic hazard assessment. Three expert elicitation workshops were held during the reporting period. These workshops provided the basis for preliminary calculations of ground motion and fault displacement hazard. Generally, the values were in the range expected, and they will be used as inputs for design and performance assessment. For human intrusion, results of evaluations of natural resources indicate that little potential exists for the presence of exploitable metallic or hydrocarbon resources within the Yucca Mountain Conceptual Controlled Area.

5.1 ALTERED-ZONE STUDY AND NEAR-FIELD ENVIRONMENT

The design and performance of the waste package and the engineered barrier system depend on the thermal, hydrological, geochemical, and geomechanical conditions that develop over time in the rock mass surrounding the emplacement drifts. The term "near-field environment" refers to the emplacement area. The altered zone is the part of the natural system that is likely to experience fundamental changes to hydrologic, mineralogic, or chemical conditions because of heating produced by emplaced waste (see Progress Report #16 (DOE 1997b), page A-101).

Altered-zone and near-field environment studies therefore focused on developing an understanding of the thermally driven processes that could occur as a result of repository loading, and on predicting the conditions that those processes will create in the vicinity of the emplacement drifts.

The major activities during the reporting period were construction and installation of instrumentation related to the upcoming Drift Scale Test. Accomplishments included placement of the concrete liner in the last segment of the drift, placement of the cast-in-place concrete invert, completion of the installation of the wing heaters, installation of hundreds of temperature sensors and other sensors, and conducting various ambient characterization activities such as air permeability testing. In addition, predictive analyses to forecast the results of the Drift Scale Test were documented in a series of reports (CRWMS M&O 1997ca; Francis et al. 1997; Buscheck et al. in prep.; Tsang and Birkholzer 1997; and Glassley 1997). Much of the field and laboratory testing conducted as part of the preheating characterization of the test domain has also been documented (CRWMS M&O 1997cb). Preparations for initiating the heating phase of the test are on schedule; heating should begin in early December 1997 and continue for up to four years, followed by a four-year cooling period.

Other efforts focused on predicting the near-field environment of the proposed repository. Wilder (in prep. [a]) described (1) conditions in the near-field environment and altered zone that may result from heating associated with emplacing waste; (2) near-field and altered zone parameter values; and (3) processes that will change the near-field and altered zone environmental conditions. Current information indicates that the geochemical processes that result from the coupling of heat and fluids generated from waste emplacement could produce large volumes of sorptive minerals within the altered zone. Furthermore, based on test results, there will likely be significant localized changes in the hydrologic system, such as flow path alteration, because of dissolution and fracture healing (Lin 1990; Lin et al. 1995; Lin and Roberts 1996). The combination of such changes could increase sorption of radionuclides and increase groundwater transport times near the repository and through the underlying part of the unsaturated zone. Such increases have potential to improve system performance. This is balanced against the potentially adverse effects of decreased travel times and bypassed sorptive materials that might result from the possible generation of large volumes of colloidal or microbial materials.

Work reported here is based on the understanding that waste package interaction with the near-field environment is sensitive to the quantity, temperature, and chemical content of both liquid water and water vapor as well as drift-gas composition (i.e., the relative concentrations of oxygen and carbon dioxide). The contact mode of water (dripping, continuous contact, or condensation) is also an important factor in the performance of the container. Therefore, this technical work focused on determining the quantity and chemistry of water that might contact the waste package. The amount of water that could contact the waste package depends on both the meteoric water percolation flux and on the flux of water that is thermally mobilized by being driven out of the pores of the rock. For the percolation flux cases analyzed in the Near Field Environment Report (Wilder in prep. [a]), the calculated estimate of flux (inflow that could contact the waste packages) is less than 1 $mm^3/m^2/yr$. However, two studies indicate the likelihood of little or no flow into the drifts at the percolation fluxes that are believed to exist at Yucca Mountain (Nitao 1997; Bodvardsson and Bandurraga 1996). For example, Nitao (1997) predicted that even after the thermal pulse no longer restricts inflow, influx into the drifts is unlikely unless percolation flux exceeds 10 mm/yr.

Preliminary results indicate that the inflowing water (i.e., transient fracture flow) that contacts (drips onto) the waste packages and interacts with emplaced materials is water that has been altered in response to repository and altered zone conditions. This alteration is expected to evolve over time (Wilder in prep. [a]). For example, an alteration of the groundwater is expected to produce solutions with moderately high pH (9 to 11) and ionic strengths greater than those of the present ambient porewater (Wilder in prep. [a]). Alteration of inflow, however, was not seen as a major determinant of waste package life, because little or no liquid water is expected to contact the waste package (Glassley 1996; Wilder 1996; Wilder in prep. [a]). Relative humidity appears to be a more significant environmental variable affecting the corrosion rate of the outer barrier of the waste package (CRWMS M&O 1997cu; Wilder in prep. [a]). Model predictions of corrosion (CRWMS M&O 1997cu) show that almost 100,000 years would be required for corrosion to remove half of the original outer barrier thickness, under assumed repository operating conditions of temperature and relative humidity. This prediction contrasts markedly with estimates of about 10,000 years for removal by corrosion under undisturbed conditions. Additional work needs to be done before making any final predictions.

Field and In situ Tests – During this reporting period, some preliminary results were obtained from the Large Block Test being conducted at Fran Ridge (Wilder in prep. [b]). The Large Block Test was designed to create controlled boundary conditions to observe and test some of the coupled thermal hydrological geomechanical and geochemical processes. Permeability measurements (Wilder in prep. [b]) indicate a range from nearly 1 darcy to as low as 1 millidarcy (approximately three orders of magnitude), with permeability closely related to structures in the block (Wilder in prep. [b]). Temperatures in the heater plane rose to more than 130°C, and refluxing appeared to be developing (Wilder in prep. [b]). After heating was initiated, a drying zone developed around all the heaters. The saturation decreased from a range of 77 to 96 percent (pretest values), to 5 to 48 percent (Wilder in prep. [b]). Thermal expansion of the block was also noted within a few hours of the beginning of the heating (Wilder in prep. [b]). For most of the block, the expansion was generally consistent with modeled continuum approaches. However, in the upper third of the block, the amount of horizontal expansion was greater than predicted using continuum assumptions (Wilder in prep. [b]). This anomaly is being investigated further.

In other thermal testing activities, the heating element for the Single Heater Test conducted in the ESF was turned off in May 1997, ending the nine-month heating phase of the test. Monitoring of the cooling phase of the test is ongoing. Results of initial analyses of heating phase measurements were documented in the Single Heater Test Status Report (CRWMS M&O 1997cs). In addition, several reports discussed the measured thermohydrologic response of the Single Heater Test (Tsang and Birkholzer 1997; Buscheck et al. 1997; and Lin in prep.), and another report presented the electrical resistivity topography measurements (Ramirez and Daily 1997).

To summarize Single Heater Test results briefly, analyses of predicted and measured temperatures indicate that conduction is the dominant mode of heat transfer, with hydrologic processes having minor effects. Water was mobilized by the heater, apparently leaving pore spaces in the rock as vapor, which then condensed in the fracture network. Some of this condensate accumulated at the end of a hydrologic test hole and was sampled several times, and a suite of chemical and isotopic analyses were performed. Initial results indicate that the condensate interacted with minerals in the fracture network prior to accumulating in the borehole. For more detailed information on the water chemistry, see Glassley and DeLoach (1997).

During FY 1998, the following activities will occur:

- Final preparation for the Drift Scale Test will be completed and will include final instrument installation, completion of pre-test characterization of the test domain, connection of all instrumentation to the automated data collection system, and conduct of ambient monitoring prior to heater turn-on in order to establish baseline conditions.

- The Near Field Models Report will discuss the models that have been developed to assess the near-field environment, the model parameters (processes, features, physical principles, etc.), and supporting data that will be addressed by these models. The report will document the status of the models and will include technical discussions of the appropriateness of the models and confidence-building activities associated with the development and use of the models.

- The chapter of the site description describing the integrated natural system responses to thermal loading will be developed.

- The heating phase of the Large Block Test will be completed, and the cooling phase will be initiated.

- The cooling phase of the Single Heater Test will be completed, and post-test characterization of the test block in the ESF will begin. This will include some disassembly of the block in order to evaluate changes in fracture mineralogy.

5.2 SITE UNSATURATED-ZONE MODELING AND SYNTHESIS

An understanding of the unsaturated-zone flow regime is needed by performance assessment and design to quantify potential seepage into drifts. Modelers of unsaturated-zone flow and unsaturated-zone transport also need this information to understand the behavior of the Paintbrush nonwelded hydrogeologic unit (PTn) (Figure 5-2).

Lateral Diversion of Flow – The ESF south ramp hydrology study was initiated to evaluate the role of faults and contrasting hydrologic properties at lithologic contacts in controlling hydrologic conditions. Of particular importance were controls on potential fast pathways through

Formal Geologic Stratigraphy (after Sawyer et al., 1994)		Hydrogeologic Units (Modified from Montazer and Wilson, 1984)	Thermal/Mechanical Units (Ortiz et al., 1985)
Qac		Alluvium	UO
Paintbrush Group	Tiva Canyon Tuff	Tiva Canyon Welded Unit TCw	TCw
	pre-Tiva Canyon bedded tuff	Paintbrush Nonwelded Unit PTn	PTn
	Yucca Mountain Tuff		
	pre-Yucca Mountain bedded tuff		
	Pah Canyon Tuff		
	pre-Pah Canyon bedded tuff		
	Topopah Spring Tuff	Topopah Spring Welded Unit TSw	TSw1
			TSw2
			TSw3
		Calico Hills Nonwelded Unit CHn	CHn1v
pre-Topopah Spring bedded tuff			CHn1z
Calico Hills Formation			CHn2z
Crater Flat Group	Prow Pass Tuff		CHn3z
			PPw
	Bullfrog Tuff	Crater Flat Unit CFu	CFun
			BFw
	Tram Tuff		CFMn
			TRw

PR17_5-2.CDR.125.PROGRESS.PR17/11-7-97

*From Sawyer et al., 1994

Figure 5-2. Comparison of the Lithostratigraphic, Hydrogeologic and Thermomechanical units of the Paintbrush Group, Calico Hills Formation, and Crater Flat Group Used at Yucca Mountain (Modified from CRWMS M&O, 1996f) the PTn. Forty-six shallow boreholes (2 m) were drilled in the walls of the ESF south ramp in the vicinity of faults to collect core samples and to monitor in situ water potential and water content. Preliminary results suggest that water may have accumulated at the contact between the TCw-PTn units, probably because of flow in the faults, and that lateral flow of water may be occurring at this contact.

A similar study commenced in the ESF north ramp to evaluate lateral diversion of downward-percolating water above the potential repository. Twenty-one shallow (2 m) boreholes were dry-drilled horizontally into rocks in the north ramp of the ESF, 18 directly into the right rib of the north ramp and three into the right rib of the Lower Paintbrush (Nonwelded) Contact Alcove. Physical properties and hydrologic conditions, required for input to detailed modeling, were measured in a stratigraphic section from the base of the TCw, down through the PTn, and into the upper vitrophyre of the Topopah Spring Tuff (Figure 5-2). These measurements indicate that at least locally, the properties and hydrologic conditions existing in the transition from the base of the moderately welded TCw into the nonwelded PTn rocks exhibit the characteristics of a permeability barrier. Altered minerals in this zone tend to increase the contrast in hydrologic properties and enhance the retardation of downward flow, resulting in conditions conducive to lateral flow (Bodvarsson et al. 1997; Flint in prep.; Robinson et al. 1997). This finding corroborates previous interpretations of permeability contrasts between these units.

Pneumatic and Aqueous Flow Paths – Drift-scale and niche (small alcove) studies were initiated in the ESF to investigate the amount of seepage that might enter repository drifts. Two niches off the ESF main drift were excavated within the potential repository host horizon about 8 m horizontally, then sealed to prevent drying caused by ventilation of the ESF. The first niche is at the location of a known fast pathway, based on chlorine-36 data (Fabryka-Martin et al., in prep.), and the second is at a location with no evidence of bomb-pulse chlorine-36. The fast-pathway niche is located near the Sundance fault and exposes both a rubble zone and relatively intact rock. Several observations within both the ESF and in a nearby borehole indicate that the effect of drying of the rock mass, caused by ventilation of the ESF, needs further investigation, as does the effectiveness of measures taken to isolate the niches from the effects of the ventilation system. Water-potential measurements were made on the walls of the fast-pathway niche in both rubble zones and zones of relatively intact rock. Initially, water potentials for rubble zones and intact rock appeared to be the same, but since then ventilation appeared to have dried both the intact rock and the rubble zones. Observations showed that the rubble zones remain wetter than the intact rock, indicating that the higher hydraulic conductivity of the rubble zones has enabled evaporated water to be replaced from the rock mass at a higher rate. Preliminary data from other studies in the ESF suggest that the drying effects of the ventilation have penetrated to a depth of approximately 3 m into the drift wall, despite some efforts to prevent drying. In addition, data from borehole USW NRG-7a (Figure 5-1) suggests that some drying effects may have extended 26 m from the ESF. The lowest instrumented section of this borehole monitored the upper part of the proposed repository host horizon, which had been slowly re-wetting (after dry drilling) since the hole was instrumented in October 1994. Since June 27, 1997, however, this section has been drying, probably in response to ESF ventilation (USGS in prep.[a]). If correct, this observation could have significant implications for repository performance. According to data, a large volume of the fracture network in the rock would dry during repository development and loading, thereby reducing the amount of water that could potentially drip onto waste packages when the repository eventually cools.

In the ESF, a borehole drilled horizontally from the Southern Ghost Dance Fault Alcove identified two splays of the Ghost Dance fault at distances of 167 m and 198 m east of the ESF main drift. In situ pneumatic monitoring in the borehole indicated that the surface barometric pressure signal propagated rapidly to the potential repository level and was relatively unattenuated along the main trace of the southern extent of the Ghost Dance fault. The residual amplitude of the surface barometric signal was about 90 percent in the main trace of the fault in the Southern Ghost Dance Fault Alcove, whereas the residual amplitude was about 70 percent in the Northern Ghost Dance Fault Alcove (and about 25 percent in surface-based boreholes unaffected by the fault (Rousseau et al. in prep.)). The pneumatic monitoring results are consistent with carbon-14 ages of gas samples collected from the fault. Gas samples were younger to the south, indicating the fault is more open to the atmosphere in this part of the ESF. The barometric and carbon-14 data indicate relatively larger permeability to air (and presumably water) along the Ghost Dance fault in the southern part of the ESF, although the permeability is relatively large all along the trace of the fault. Subsequent excavation revealed that the splay at 167 m was the main trace of the fault.

Fast Pathways – Completion of the ESF main drift and south ramp allowed collection and analysis of samples for chlorine-36, to investigate the potential for fast pathways in the southern part of the ESF. It was also possible to make comparisons between the northern and southern parts of the potential repository area. Bomb-pulse chlorine-36 was detected in some samples taken from the north ramp and the northern part of the main drift, indicating that fast flow paths exist. Similarly, tritium was detected in water distilled from core in about half the samples from the Northern and Southern Ghost Dance Fault Alcoves, again indicating the presence of fast flow paths. These flow paths apparently have the capability to transmit water from the surface to depths of at least 300 meters in less than 50 years. Samples with elevated levels of chlorine-36 generally correspond spatially to faults mapped at the surface and in the ESF; however, there appears to be no correlation with fault type, offset, orientation, or related fracture density (Sweetkind et al. 1997a). In contrast, no samples containing unambiguous levels of bomb-pulse chlorine-36 or tritium were collected from the vicinity of faults in the southern part of the ESF. This indicates that faulting is not the only control on the distribution of fast flow paths. It appears that there are multiple, interrelated controls, including the presence of faults, sufficient net infiltration, and thickness of alluvium (Fabryka-Martin et al. in prep; Sweetkind et al. 1997a). Preliminary analysis of sample locations in the southern ESF suggests that these locations generally correspond to areas of low net infiltration, probably related to thick alluvial cover as depicted by Flint et al. (in prep.).

Moisture and Percolation at the Repository Level – Studies of secondary mineralization in fractures and subsurface cavities continued as a means to evaluate past unsaturated-zone hydrologic conditions, provide data to refine the site-scale unsaturated-zone flow model, and constrain estimates of percolation flux through the potential repository block (Paces et al. 1997b). Secondary minerals indicating connected fracture pathways were most abundant in the northern ESF, in the vicinity of Drill Hole Wash, and became scarce in the southern half of the main drift, where lower infiltration rates had been modeled in the overlying surface material. These changes in fracture connectivity are consistent with the chlorine-36 preliminary results discussed previously. Both uranium-lead and uranium-thorium data indicate that average mineral growth rates within the potential repository host horizon (TSw) are low (~1 mm/10^6 years). Rates remained remarkably constant throughout the late Tertiary and Quaternary periods. The constancy of mineral deposition rates within the TSw suggests that the overlying PTn has diminished the effects of variable infiltration caused by climate change during the past 8 million years. Additional uranium-thorium isotope data, from the outermost surfaces of minerals collected from the TCw in the south ramp, support interpretation of a difference in initial uranium-isotope ratios in minerals above and below the PTn, indicating that the water has been modified by rock-water interaction in the PTn. Porewater strontium isotopic data indicate similar modifications of water in the PTn, probably because of the transition from rapid fracture flow in the TCw to much slower, dominantly matrix flow in the PTn. Both subsurface mineral and porewater strontium-isotope data support a conceptual model in which percolation in the TSw is not necessarily directly proportional to surface infiltration because of matrix flow (and possibly lateral flow) within the PTn (Paces et al. 1997a).

Preliminary analysis of the ESF moisture-balance data indicates that evaporation rates remain fairly constant, even though water use in the ESF was highly variable. As atmospheric air first enters the ESF, the average potential evaporation rate was approximately 6 mm/day, but decreased as the humidity increased along the length of the tunnel because of a decrease in the vapor-pressure deficit. If correct, the estimate indicates daily losses from evaporation are the same order of magnitude as the annual percolation rate. The estimate shows the potential for significant drying of the repository host rocks during construction and operation of a repository, because of the existence of forced ventilation.

Unsaturated-zone hydrochemistry studies focused on reducing the uncertainty associated with apparent differences in source and age of porewater and gas collected from boreholes from above, below, and within the basal vitrophyre of the TSw hydrogeologic unit (Figure 5-2) (Yang et al. in prep.). Deuterium values for porewater from units above the basal vitrophyre indicate a warm (post-glacial) origin for the water. Deuterium values from porewater from the basal vitrophyre and from water from the saturated zone indicate a colder, glacial (and therefore older) origin. However, porewater from the Calico Hills Formation below the vitrophyre had deuterium values indicating a warm (young) origin similar to porewater from above the vitrophyre. In addition, although carbon-14 ages of gas in several boreholes increased downward within the TSw to values of 13,000 years (about the end of the last glacial period) near the basal vitrophyre (Yang et al. in prep.), carbon-14 ages of gas in the Calico Hills Formation below the vitrophyre were 6,000 years or less. Taken together, the deuterium and carbon-14 data indicate that younger water and gas enter the Calico Hills Formation under the site area by way of lateral flow and/or fast pathways that penetrate the basal vitrophyre of the TSw hydrogeologic unit (Yang et al. 1996).

Efforts continued to determine percolation flux through the unsaturated zone using site data and techniques that are independent from the numerical infiltration model (Flint et al., in prep.), and the site-scale unsaturated-zone flow model (Bodvarsson et al. 1977). One technique used borehole temperature profiles to estimate percolation flux by varying the assumed infiltration rates in multiple one-dimensional, coupled heat and liquid flow models (TOUGH2), until an optimal match between the simulated and observed borehole temperature profiles was obtained. Estimates of percolation flux for ten borehole locations ranged from 0.5 mm/yr in a wash to 20 mm/yr on central Yucca Crest. The estimates of percolation flux were compared with infiltration rates that were estimated by numerical infiltration modeling on the basis of a soil water-budget model (Flint et al. in prep.). Percolation fluxes estimated from the temperature data produced higher estimated infiltration rates at most locations. Specifically, estimates of percolation flux based on borehole temperature profiles were consistent with the relatively high infiltration fluxes estimated for Yucca Crest, but the results indicate that net infiltration in some washes may be greater than currently predicted by the numerical infiltration model (Flint et al. in prep.).

Another technique for estimating percolation flux through the unsaturated zone used a chloride mass-balance approach and chloride-concentration data for water samples taken from the rock matrix, perched-water, and the uppermost part of the saturated zone. The porewater extracted from the rock matrix, in addition to being analyzed for chloride, was analyzed for bromide (Fabryka-Martin et al. in prep). Based on the low bromide to chloride ratios, this water had not been influenced by ESF construction water (which has a bromide tracer added to it). The basic chloride mass-balance model (which assumed well-mixed water) was adapted for use in dual-permeability environments in which the matrix and fracture water may remain incompletely mixed. The adapted model was then applied to Yucca Mountain using average chloride concentrations for the PTn, the Calico Hills Formation, and perched water. The average site-wide percolation rate estimated using this chloride mass-balance technique was 9.8 mm/yr. Only about 10 percent of this flux was determined to have occurred as matrix flow through the PTn and Calico Hills; the remainder of the flux in these simulations was determined to have been transmitted through these units as fracture flow. The chloride mass-balance approach estimate is larger than the percolation flux estimate of 6.7 mm/yr used in the unsaturated-zone model study (Bodvarsson et al. 1977).

The different techniques used for estimating percolation flux (some of which have been discussed here) were not directly comparable. Some researchers estimated flux at a specific location, such as a borehole, while others produced site-wide average estimates. The various methods also rely on different physical and chemical measurements and assumptions. By using and evaluating multiple techniques, and collecting numerous types of data (for example, matrix hydrologic properties), the Project continued to improve and refine estimates of percolation flux, which is a key parameter in performance assessment calculations.

Flow and Transport Modeling – The site-scale unsaturated-zone flow model (Bodvarsson et al. 1997) was used to evaluate the uncertainty associated with several key aspects of flow in the unsaturated zone at Yucca Mountain. These included flow patterns and percolation flux based on geochemical data, hydrologic properties of fractures, conceptual models of fracture flow and fracture-matrix interaction, and potential moisture seepage into drifts. The uncertainty associated with percolation flux at the repository horizon was reduced by simulations that demonstrated that average percolation-flux values exceeding 20 mm/yr are inconsistent with available temperature, chloride, and matrix-property data. The simulations also demonstrated that the lower limit of average percolation flux is about 0.5 mm/yr, which accounts for evaporative processes associated with the high gas permeability of the TCw unit and the relatively high saturations in some of the welded hydrogeologic units. Simulations using the ambient drift-scale unsaturated-zone model (Tsang et al. in prep.) indicate that drift seepage in heterogeneous media seems to be controlled by channeled flow and localized ponding. Further, modeling the system using homogeneous properties resulted in estimates of percolation fluxes that caused seepage into drifts to be several orders of magnitude too great. Therefore, simulation of heterogeneity is a key factor for performance assessment.

The site-scale unsaturated-zone transport model (Robinson et al. 1997) evaluated many different types of conceptual models and numerical methods to simulate the known distribution of environmental isotopes and water chemistry in the unsaturated zone. Simulated travel times in the unsaturated zone varied widely depending on the selection of parameter sets, but generally showed distinct differences from south to north under the potential repository, consistent with the data being collected from the completed ESF. Variability in travel times ranged from 10 years to 10,000 years using three-dimensional simulations. This study found that, if fracture flow is the predominant mechanism and water can contact the waste form, radionuclides could reach the water table quickly. The conclusion depends on the existence of either of two conditions: (1) if the Calico Hills nonwelded unit (CHn) (Figure 5-2) were fractured and radionuclides did not come in contact with the zeolites, or (2) if colloid transport were significant for certain highly sorbing radionuclides such as plutonium (see Section 5.4). An investigation of small-scale variability of chemical and hydrological properties was done using geostatistical interpretations rather than cutoff values for defining model layers. A key conclusion is that the retardation calculated using this method is much larger than the retardation estimated using a zeolite threshold method (which causes much of the zeolites to be bypassed by forced fracture flow). In addition, the results suggest that using a small, but non-zero distribution coefficient for vitric units may be an effective way to mimic more rigorous conditional simulations with the site-scale model.

Abstraction of the unsaturated-zone flow model is described in Section 3.2.3.

During FY 1998, the following activities will occur:

- The magnitude of percolation flux through the unsaturated zone and the locations of recharge to the saturated zone will be investigated through analysis of hydrochemical data, which will be collected from the uppermost part of the saturated zone. This work will be part of the renewed groundwater sampling activities.

- Collection of moisture balance data will be initiated in the East-West cross drift starter tunnel to be excavated for the enhanced characterization of the repository block. Monitoring of water use in the East-West cross drift will be supplemented with fixed-location sensors and sensors installed in the ventilation line at the tunnel boring machine as part of the moisture balance study. Neutron logging, geochemical analysis of core, and borehole instrumentation will be combined with the moisture-balance study to evaluate the movement and exchange of construction water and groundwater.

- Distribution of chloride and chlorine-36 in the cross drift will be predicted, based on current conceptual and numerical flow and transport models for the unsaturated zone. Chloride, bromide, and chlorine-36 concentrations in porewater extracted from drill core from the ESF will be analyzed to provide a surrogate measure of infiltration rates, evaluate the role of the PTn in moderating downward fluxes, and monitor the extent to which construction water has migrated into the floor and walls of the ESF.

- A detailed, two-dimensional, cross-sectioned, numerical model of the ESF north ramp will be developed to investigate the potential for lateral diversion of downward-percolating water by the hydrologically significant stratigraphic region near the top of the PTn. The model will incorporate information from vertically drilled boreholes in the Upper and Lower Paintbrush (Nonwelded) contact alcoves. After the detailed hydrological-properties database has been expanded sufficiently, the three-dimensional unsaturated-zone, site-scale flow model will be used to perform simulations at a larger scale to investigate the potential for lateral diversion of water by the PTn.

- Additional instrumentation will be installed in the fast-pathway niche. Water-potential sensors will be installed around the niche to determine if increases in saturation occur from percolating water. Borehole instrumentation strings will be installed in the five boreholes drilled radially from the niche. The instrumentation will be used to investigate moisture conditions in the local fracture networks and the high water-content anomalies near the Sundance fault. The non-fast-pathway niche will be outfitted with water-potential, temperature, and relative-humidity sensors. Two additional niches will be excavated and outfitted with instruments in the ESF main drift, one in the intensely fractured zone and one under the alcove near where the East-West cross drift will cross over the main drift.

- In support of performance assessment flow-model abstractions, the importance and spatial distribution of short trace-length fractures in the ESF will be evaluated. Studies of fast-flow paths will continue in the ESF by investigating the correlation of structural features and chlorine-36 occurrences. To provide information on the influence of faults on the fracture network within the potential repository host horizon, a fracture study will be conducted near a large fault at Busted Butte, where these rocks are exposed at the surface.

- Porewater from core obtained from new boreholes USW WT-24 and USW SD-6 (Figure 5-1) will be analyzed, and the data will be added to the existing database for porewater ages, compositions, and nature of flow. Gas samples from existing boreholes will be dated by carbon-14 methods and analyzed for carbon-13 to improve understanding of vapor-phase transport and liquid-gas equilibrium between porewater, fracture water, and gas in the repository block.

5.3 SATURATED-ZONE HYDROLOGIC SYSTEM MODELING AND SYNTHESIS

Studies of the saturated zone, in particular developing data and predictive models that improve understanding of potential flow paths and travel times of groundwater and radionuclides, are needed to support performance assessment calculations. During this reporting period, studies focused on reducing uncertainties about the flow regime beneath the potential repository, determining how potential fast pathways might impact radionuclide travel times, determining the potential use of existing water chemistry and isotope data to bound the range of past flow fields, and (with the initiation of borehole USW WT-24) determining the cause of the large hydraulic gradient northwest of the site (Figure 5-1).

Saturated Zone Modeling – Modeling efforts this reporting period addressed flow in the saturated zone at three levels of detail: regional-scale, site-scale, and sub-site-scale.

Regional modeling of the saturated zone investigated simulated effects of climate change on the regional flow system (D'Agnese et al. in prep.). The regional model provides a basis for further consolidation with the NTS model (IT Corporation in prep.), as well as incorporation of boundary conditions for the site-scale model (D'Agnese et al. 1997). Geologic data from the Integrated Site Model (ISM2.0) and recently constructed geologic sections, provided by the DOE Nevada Operations Office for the NTS and vicinity (IT Corporation in prep.), were incorporated into a site-scale model update featuring a refined grid spacing, an improved representation of the geometry of hydrogeologic units, and a more realistic distribution of hydraulic properties. The model update allows a more accurate calculation of ground-water flow paths and fluxes (Czarnecki et al. in prep.). Differences between simulated and observed water levels of less than 10 m for the majority of model simulations which indicate reasonable calibration of the flow model was achieved. However, significant uncertainty remains in fluxes beneath the repository area, because fluxes calculated by the site-scale model were about twice those calculated by the regional flow model at the southern boundary of the model and 10 times greater in the northern areas of the site model. Nevertheless, the flow model calibration did reduce the uncertainty by validating the existence of three conceptual aspects of the site-scale flow system: groundwater recharge in upper Fortymile Wash, a low-permeability barrier south of the large hydraulic gradient, and a low-permeability barrier along Solitario Canyon.

An updated three-dimensional flow and transport model for the site-scale saturated-zone flow model was developed (USGS in prep. [b]; Zyvoloski et. al. 1997). Transport modeling results captured the two end-member flow modes (fracture-dominated and pervasive matrix diffusion) well, but failed to produce realistic results for small but non-negligible diffusion into the rock matrix.

In addition to site and regional-scale modeling activities, studies using a sub-site-scale saturated-zone hydrologic model were conducted. The studies were to evaluate the effects of the large hydraulic gradient, of flow directly below the repository, and of impact of structural features such as fault zones. They also support preparatory work for the second testing complex (Cohen et al. 1997).

Saturated-Zone Flow Rates – Existing saturated-zone hydrochemical and isotopic analyses from all wells in the immediate vicinity of Yucca Mountain were synthesized into an integrated database that was used to form a conceptual model of saturated-zone flow paths, including areas of likely recharge in the vicinity of Yucca Mountain. Age correction, using the interactive code NETPATH (Yang et al. in prep.; Plummer et al. 1994), and analysis of existing carbon-14 data indicate that saturated-zone water beneath Yucca Mountain is about 7,000 years old, younger than previously believed. Further, water in the saturated zone beneath Fortymile Wash (Figure 5-1) appears to be about 4,000 years old, and water in the saturated zone further to the south of Yucca Mountain appears to be about 12,000 years old. These results however, are not consistent with the deuterium data discussed in Section 5.2, and will be further evaluated. Mass-balance analysis of chloride concentrations in saturated-zone water indicates that the water beneath the potential repository block is derived locally or from the north rather than from the west. The significance of this result is being evaluated.

The collection of data for calculation of ground-water travel times and solute-transport rates in the saturated zone continued through ongoing hydraulic and tracer tests at the C-hole complex about 3 km southeast and downgradient from the potential repository (Figure 5-1). A long-term hydraulic test has been underway at the C-hole complex since May 8, 1996, in the Lower Bullfrog interval. Analysis of drawdown data from all the observation wells indicate that the Miocene tuffaceous rocks being tested act as a single highly transmissive aquifer. For detailed information on the characteristics of the aquifer, see Geldon et al. (in prep.)

Superimposed on the long-term hydraulic test at the C-hole complex (Figure 5-1), two conservative-tracer tests with radially convergent flow fields toward the pumped well (UE-25 c#3) were initiated on January 9 and 10, 1997, by injecting a chemical tracer (Pyridone) into hole UE-25 c#1 and a different tracer (DFBA) into hole UE-25 c#2 (See Site Characterization Progress Report: Yucca Mountain, Nevada, # 16 (DOE 1997b), Section 3.1.14). Analysis of DFBA results yielded information on the following parameters important to modeling flow and transport: flow-path porosity (which is composed of fracture and matrix segments), storage porosity, and longitudinal dispersivity (Geldon et al. in prep.). Analysis of the Pyridone breakthrough curve was not performed, because arrival of the peak concentration of Pyridone had not yet occurred.

Borehole USW WT-24 (Figure 5-1) is being drilled and was tested in an attempt to reduce the uncertainty associated with the postulated existence of a large hydraulic gradient north of the site (Figure 5-1), and to determine whether or not perched water exits in this area. This information is needed to reduce the uncertainty in the site-scale, saturated-zone flow model with respect to these major hydrologic features, so that more credible flow-rate estimates can be made. Drilling of USW WT-24 began on July 23, 1997. As of September 30, 1997, the borehole was completed to a depth of 473 meters. The final depth of borehole USW WT-24 is expected to be approximately 884 meters.

During FY 1998, the following activities will occur:

- The C-hole complex will be reconfigured to conduct hydraulic and tracer tests in the Prow Pass Tuff. This unit is the uppermost unit in the saturated zone directly beneath the potential repository and, therefore, is one of the first saturated units that would be encountered by any radionuclides that might migrate from the repository. Hydraulic and transport properties of the low-transmissivity Prow Pass Tuff will augment properties already determined for the underlying highly-transmissive Bullfrog Tuff and provide a full range of values for use in performance-assessment calculations.

- The saturated-zone hydrochemical database will be expanded to include existing data collected further from the repository block. The exact format will be determined by a consensus of all potential users involved in the Project. Extensive water sampling and analysis will be conducted for new boreholes USW WT-24 and USW SD-6 and for existing boreholes UE-25 WT#3 and UE-25 WT#17 (Figure 5-1).

- Additional work for the regional model will focus on improving estimates of discharge from regional springs and evapotranspiration areas, in order to reduce uncertainty in regional model results. The site-scale model will be further refined to include explicitly hydrologically significant faults. The flow model will incorporate the updated hydrogeologic framework model, and will contain a refined finite-element mesh. Temperature and hydrochemical data and information from testing at the C-hole complex will be used, to the extent practical, to aid in the calibration process. Work will continue on testing alternative conceptual models of the large hydraulic gradient with the flow model.

5.4 SITE GEOCHEMICAL SYNTHESIS AND MODELING

Geochemical investigations and resulting data contribute to a variety of Project information needs in design, performance assessment, flow and transport modeling, and mineral alteration studies. Work during this reporting period concerned four research topics: analysis of plutonium in groundwater data from the Nevada Test Site Environmental Program (NTS-ER) studies, and laboratory studies of neptunium, Yucca Mountain groundwater chemistry, and batch sorption studies.

Colloidal Transport Studies – During this reporting period, NTS-ER reported that plutonium was discovered to be associated with colloids in NTS groundwater (Thompson et al. in prep.; Kersting et al. in prep.). This is the first documented transport of plutonium with natural colloids at the NTS. Plutonium occurred in volcanic rocks similar to those at Yucca Mountain. Colloidal transport of plutonium and other strongly sorbing radionuclides may become important in performance assessment calculations. The rationale for investigating radionuclide transport via colloids at Yucca Mountain is described in the SCP (Activity 3.8.1.3.5.2.1). Furthermore, the potential for colloidal transport of radionuclides affects both the design of and materials used in the engineered barrier system.

The plutonium is from a 1968 nuclear test located on the NTS, about 32 km north of Yucca Mountain. Based on the analysis of available well samples, a few femtocuries (10^{-15} curies) of plutonium migrated about 1 km in the 29 years since the test. The majority of the plutonium detected by NTS-ER was associated with the colloidal fraction and was not present in the dissolved (ultrafiltrate) fraction. The colloidal material associated with the plutonium consists mainly of clays (illite), zeolites (mordenite) and cristobalite. All these minerals represent phases derived from the alteration and weathering of the local rhyolitic tuff.

Related laboratory studies of colloidal transport (Conca et al. 1997) indicate that colloidal charge, not size, is the major controlling factor in the transport of colloids in unsaturated tuffs. Colloid stability and concentration increase in a given groundwater, when the ionic strength of the host groundwater decreases and/or the organic content of the host groundwater increases (Triay et al. 1997a). Colloid concentrations in waters from well UE-25 J#13 were on the order of 10^6 particles/ml (Degueldre et al. 1996; Triay et al. 1996). At this low particle loading (Triay et al. 1997a), the sorption of radionuclides onto natural colloids would have to be very high (Ogard 1987) and irreversible (Triay et al. 1997a), before significant quantities of plutonium could be carried (Robinson et al. 1997).

Laboratory Studies of Neptunium – Studies of neptunium are significant to the Project because this long-lived radionuclide has considerable impact on performance assessment calculations. Of particular importance is the solubility of the various species of neptunium (Np). For example, the finding of a Np(IV) solid as the solubility-limiting precipitate would allow the Project to decrease the solubility of neptunium by several orders of magnitude in the total system performance assessment. Although geochemical modeling using the current YMP GEMBOCHS database indicates that Np(IV) solids should form, high temperature kinetic tests to date indicate no evidence of formation of Np(IV) solid in either crystalline or amorphous phases. (Note: Project chemists have described other Np solid precipitates using a variety of techniques; see Efurd et al. 1996; 1997, and Runde et al. 1997). Current research is investigating the solubility product (equilibrium value) at constant ionic strength on the solubility-limiting Np(IV) solid (Runde et al. 1997). This work has yielded some data that supports reducing neptunium solubilities from those used in previous performance assessment calculations.

Groundwater Chemistry at Yucca Mountain – Two main types of water have been described at Yucca Mountain. Type-1 waters occur as porewater in the hydrologic units above the Calico Hills Formation. These waters have a higher ionic strength than Type-2 waters, and their composition is strongly influenced by soil zone processes such as dissolution and precipitation. Type-2 waters occur as perched and saturated-zone groundwaters. These waters have low ionic strengths and are affected by hydrolysis reactions that involve carbonic acid (Meijer 1994; Yang et al. 1996; in prep.). Porewater in the Calico Hills Formation appears to be a mixture of Type-1 and Type-2 waters (Meijer 1994, Yang et al. in prep.). Chemical data indicate that porewater comes to equilibrium very slowly (Meijer 1994; Yang et al. 1996; in prep.). Therefore, future variations in the porewater chemistry at Yucca Mountain under ambient conditions are likely to be minor (Robinson et al. 1997). An update of the groundwater chemistry model (Meijer 1996) is presented in Robinson et al. (1997) and Bodvarsson et al. (1997). These updated models, which use the current environmental isotope and groundwater chemistry data, confirm that the compositions of UE-25 J#13 and UE-25 p#1 waters appear to bound the composition of groundwater seen at Yucca Mountain. Furthermore, these models indicate that the flow and residence times of groundwater in the various units at Yucca Mountain are consistent with the measured hydrogeochemical field data.

Work in support of transport modeling focused on the carbonate ion, a very strong complexing agent for radionuclides, which occurs in significant concentrations in Yucca Mountain groundwater (Meijer 1994). In addition, both neptunium and plutonium (Pu) can hydrolyze water (Liester and Muhlenweg 1988; Runde et al. 1996), making hydroxyl complexes potentially important in the Yucca Mountain system. Results of modeling the potential stability of hydroxyl, carbonate, and mixed hydroxyl-carbonate complexes in the Yucca Mountain saturated zone waters using the EQ3/E6 code, suggest the N_pO_2+ and $N_pO_2CO_3^-$ are the dominant species containing Np(V) (Runde et al. 1997; Efurd et al. 1997) and that $Pu(OH)_5^-$ is the dominant species with PuO^2 or $Pu(OH)_4$ solids (Triay et al. 1997a). The bulk solubility of plutonium was measured in water collected from UE-25 J#13, and the solubility has a narrow range from $4x10^{-9}$ to $5x10^{-8}$, and Pu(IV) is the solubility-controlling solid (Triay et al. 1997a). This work is being coordinated with the neptunium solubility studies.

Batch Sorption Studies – Results of laboratory studies (Conca et al. 1997; Triay et al. 1997b) indicate that as the degree of saturation decreases in Yucca Mountain tuffs, the transport time for radionuclides increases. This increase in transport time with decreased saturation occurs because transport only occurs in the fluids (Triay et al. 1997b); as the saturation decreases, there is less fluid available for transport. Furthermore, with reduced saturations in the unsaturated zone, the fluid-rock interaction time is increased (Triay et al. 1997b), and there is a concurrent increase in the potential for sorption (Triay et al. 1997a; b). Therefore, radionuclide transport is potentially much slower under drier unsaturated conditions (Triay et al. 1997a). Based on water-potential measurements in the ESF and USW NRG-7a discussed in Section 5.2, the development of a relatively dry repository appears feasible.

Laboratory experiments indicate that sorption of anions appears to be negligible in all Yucca Mountain rocks, except those containing zeolites (Triay et al. 1997a; b), where small but measurable amounts of anion sorption can be detected. Thus the retardation of anions could be minor in the unsaturated zone, and their transport would be controlled by movement of the fluid phase containing them. If correct, this interpretation indicates that the unsaturated zone away from the effects of the engineered barrier system may be less effective in retarding elements that form anions (Bodvarsson et al. 1977; Robinson et al. 1997). As suggested above, however, a drier repository may partially compensate for negligible anion sorption.

During FY 1998, the following activities will occur:

- A report will be developed summarizing the occurrences of radionuclide migration via colloids both at the NTS and elsewhere in the world, and the NTS occurrences will be assessed for similarities to Yucca Mountain conditions. Performance assessment calculations will be performed to estimate colloidal mobility of plutonium at Yucca Mountain, and this information will be compared with existing information. Planned laboratory studies now include work on the formation and stability of plutonium colloids; the reversibility of plutonium sorption onto iron oxide, clay, and silica colloids; and the transmission of colloids through unsaturated media using the unsaturated flow apparatus.

- In addition, samples will be taken from various saturated-zone wells to access and bound the compositions and quantities of colloids in both saturated-zone and perched groundwater at Yucca Mountain. Based on these studies, a recommendation will be made to performance assessment about the need to include colloids in future calculations.

- Higher temperatures and oxidation-reduction-active co-precipitates, which may be relevant to Yucca Mountain, will be used in experiments designed to produce Np(IV) solids and to determine whether such solids will be stable under Yucca Mountain repository conditions.

- A field test of the unsaturated-zone transport properties of the Calico Hills Formation is planned to address uncertainties concerning the transport of neptunium, technetium, iodine and colloid-bearing radionuclides (mainly plutonium) through both fractures and the matrix. Construction of the testing facility in the Calico Hills Formation at Busted Butte is underway, and the design, instrumentation, and tracer analysis of a test bed has been funded.

5.5 INTEGRATED SITE MODEL

The Integrated Site Model is being developed to provide the combined geologic rock properties and mineralogic framework to support hydrologic and transport models of Yucca Mountain. The models are used to evaluate technical issues including flow pathways, radionuclide transport, and other parameters of the host rock system.

Reevaluation of Lithostratigraphic Contacts and Site Structures – During this reporting period, efforts concentrated on updating the geologic framework of the Integrated Site Model by reinterpreting key subsurface lithostratigraphic contacts. The primary information source for the reinterpretation was the Project's quality-assured, borehole geophysical database. This activity standardized the identification criteria for lithostratigraphic units, resulted in qualification of most of the data, and allowed inclusion of previously-acquired data in current interpretations. The revised database provided increased consistency in locating lithostratigraphic contacts, reduced uncertainty in lateral correlations of units, and contributed directly to the development of a qualified three-dimensional geologic framework for the Integrated Site Model. The 1:24,000 scale site area geologic map with accompanying text (Day et al. 1997) was completed. (Note: This work will also provide key input to the geology chapter of the working draft license application.) The mapping resulted in new insights into the evolution and spatial variation of site structures. Faults at Yucca Mountain were classified as block-bounding (hundreds of meters of displacement), relay (which transfer displacement between block-bounding faults), and intrablock (short, small displacement faults that accommodate intrablock deformation). This hierarchy of faults is captured in the Integrated Site Model. In addition, a structural model of the Paintbrush Group was developed (Potter et al. 1997). Potter et al. (1997) found that a gradation of intrablock faults exists from those that inherited their near-vertical dips and discontinuous nature from cooling joints along which they formed (e.g., Sundance fault) to those whose geometry appears to be controlled by block-bounding faults (e.g., Ghost Dance fault). The stratigraphic control of jointing so important to intrablock fault growth did not affect the propagation and geometry of block-bounding faults, which reflect regional tectonic influences.

ESF Geologic Mapping – During this reporting period, geologic mapping of the ESF was accomplished all the way to the south portal, thereby completing this activity. Underground geologic mapping was also completed for the northern Ghost Dance Fault alcove. Analysis of the ESF mapping for the southern part of the main drift and the south ramp indicate that the stratigraphy and structure are very consistent with preconstruction predictions (Eatman et al. in prep.). These important results provide a basis for validating this part of the Integrated Site Model and generally improve confidence in the accuracy of the model. Also, geologic mapping of the Thermal Test Facility, where the Drift Scale Test is located, was completed. The mapping confirmed that the rock in the Thermal Test Facility is nearly identical to that in the main drift, and is in the upper part of the potential Repository Host Horizon.

All quality-assured fracture data from surface studies, the ESF, and boreholes were synthesized into a single data set (Sweetkind et al. 1997b). Although borehole fracture data are not directly comparable to surface and ESF data, data from each source showed common trends in fracture intensity for each of the model units. Control of fracture attributes (type, spacing, continuity, number of sets, and probably connectivity) appears to be more closely related to lithologic characteristics than to structural influences. The synthesis is important because it contributes directly to the development of a qualified unsaturated-zone flow model.

Geophysical Studies – In an effort to better characterize the Tertiary-Paleozoic interface in the Integrated Site Model, vertical seismic profiling data, which had been acquired in the UE-25 UZ#16 borehole (Figure 5-1), was reinterpreted (Feighner et al. in prep.). The deeper reflections observed in the data could indicate the Paleozoic surface; this preliminary interpretation will be revisited following obtaining the results of the vertical seismic profiling survey to be conducted in borehole UE-25 p#1 (Figure 5-1).

During FY 1998, the following activities will occur:

- The revised borehole geologic data and recent geologic mapping will be incorporated into the Integrated Site Model. The updated geologic framework for rock properties and mineralogical models will be updated to incorporate new data, integrate geophysical data with core measurements, and provide a standardized method of data extraction. The model's predictive capabilities and sources of uncertainty will be assessed. New data and improved modeling techniques are expected to improve the ability for the Integrated Site Model to predict the subsurface locations of faults and rock unit boundaries throughout the Yucca Mountain site. The revised borehole data are also expected to reduce uncertainty in modeling stratigraphic lateral variability in

Paintbrush Group rocks. The Integrated Site Model will be tested using data provided by boreholes drilled in FY 1998. Several studies are planned for the ESF including completion of mapping in the southern Ghost Dance Fault access drift and initiation of mapping in the east-west drift. A predictive report on the structural geology of the East-West cross drift will also be prepared, in part to assess further the predictive capabilities of the Integrated Site Model. (See Section 4.4 for additional information about the East-West cross drift.) Compilation of a 1:50,000-scale geologic map will begin in support of the site saturated-zone flow model.

- Vertical seismic profiling surveys will be conducted in borehole UE-25 p#1 (Figure 5-1) to characterize the Tertiary-Paleozoic contact. The results will apply to reassessing vertical seismic profiling data acquired in other boreholes that will better define the Tertiary-Paleozoic contact across the site, and to calibrate the gravity modeling of the Tertiary-Paleozoic surface. The refined surface will be used as input for the next revision of the Integrated Site Model, ISM3.0.

5.6 THREE-DIMENSIONAL ROCK PROPERTIES SYNTHESIS AND MODELING

The physical properties of a rock or rock unit (such as density, porosity, or thermal conductivity), have a direct effect on how specific processes (such as groundwater flow) will occur in that rock unit. Therefore, in order for a process model to produce representative results, it must be run on a framework that provides realistic physical properties and geometries for the rock units. The development of a three-dimensional rock properties model provides critical support to all unsaturated-zone, saturated-zone, and thermal process modeling efforts; and ultimately to total system performance assessment calculations. The rock properties modeling effort is coordinated with, and becomes part, of the Integrated Site Model discussed in Section 5.5.

During this reporting period, a report was submitted that describes the three-dimensional hydrological and thermal property models of Yucca Mountain (Rautman and McKenna 1997). Geostatistical methods were used to simulate the properties of three geologic units: the nonwelded and principally vitric part of the upper Paintbrush Group (PTn model unit); the densely welded and principally devitrified rocks of the Topopah Spring Tuff (TSw model unit); and the nonwelded to partially welded and variably zeolitized units of the Calico Hills Formation and Prow Pass Tuff (the Ch-PP model unit). The rock properties modeled include porosity, bulk density, and saturated hydraulic conductivity for each unit, and thermal conductivity for the TSw model unit.

The modeling successfully used data from a diverse set of site characterization measurements to provide the first laterally extensive, site-scale, fully three-dimensional model of material properties at the Yucca Mountain site. The model indicated substantial material-property heterogeneity, both vertically and laterally, and provided the basis for quantitative estimates of the spatially variable geologic uncertainty. The modeling also identified several differences in data from different sources.

During FY 1998, three-dimensional rock properties models will be updated to incorporate the latest geologic framework developed for ISM 3.0 additional hydrologic properties data from boreholes and the ESF, and to integrate sample measurements with properties calculated from geophysical logs (see Section 5.5). Models will be developed for the PTn, TSw, and CHn thermomechanical units (Figure 5-2). Properties that will be modeled include porosity, density, thermal conductivity, and saturated hydraulic conductivity. These data will be incorporated into the development of ISM 3.0, which is planned for completion in FY 1999.

5.7 BIOSPHERE, METEOROLOGY, AND PALEOCLIMATE AND PALEO-ENVIRONMENTAL SYNTHESIS

Studies of current climate and weather (meteorology) are required to estimate infiltration into the unsaturated zone (information needed by design and performance assessment), as well as to evaluate atmospheric dispersion and airflow, which are inputs to dose assessments for preclosure radiation safety analyses. Paleoclimate and paleoenvironmental data are needed to bound past climatic fluctuations (particularly variability of precipitation and temperature) and to aid in estimating previous infiltration into the unsaturated zone, and impacts on water table elevation and the groundwater flow regime. This information is then used to estimate the range of changes in hydrologic conditions related to anticipated future climate changes.

Biosphere – Preliminary Biosphere Dose Conversion Factors were calculated to support TSPA-VA. Studies assessed how radionuclides potentially released from the proposed repository could be transported through a variety of environmental media (pathways) to humans and calculated the radiation dose. Sensitivity analyses were performed to identify the most important pathway parameters, and a regional survey was conducted to characterize the critical exposure group for the modeling effort. The Biosphere Dose Conversion Factor of a radionuclide is the annual radiation dose resulting from unit radionuclide concentration in groundwater under a specific exposure scenario, i.e., mrem/yr per pCi/L. The potential radiation dose from the repository can be calculated by multiplying the radionuclide concentrations in groundwater (pCi/L) and the Biosphere Dose Conversion Factors. This calculation is one of the bases for evaluating the risk to humans represented by the repository.

Meteorology – Two milestone reports on meteorology were completed during this reporting period. Weather-related information supporting engineering design and presenting local and regional climate was presented in CRWMS M&O, 1997ct. Airflow and atmospheric dispersion characteristics were presented in CRWMS M&O, in prep.[j]. In addition, a direct comparison of data from the National Weather Service stations, located near Yucca Mountain and having long-term records, with data from the YMP stations (collected since 1985), indicated that the National Weather Service data could be used as long-term analogs for site meteorological data. This analog relationship reduces the uncertainty of using the long-term offsite records to describe conditions at Yucca Mountain over a longer period of time.

Paleoclimate and Paleoenvironmental Synthesis – Analysis of the extensive data set documenting past climate conditions within the Yucca Mountain region indicates that past climates changed from interglacial conditions to glacial conditions in less than 1,000 years

(Bradbury 1997a). Furthermore, changes within a glacial or interglacial period (wetter to drier for example) occurred in less than 300 years. Interpretation of paleoclimate data also suggests that the long-term average levels of effective moisture during the past 400,000 years were higher than today about 80 percent of the time (Bradbury 1997b), because of greater precipitation, less evaporation, or both. Among the four major glacial periods during the past 400,000 years, the last two (170-130 ka and 40-12 ka) appear to have been much colder than the first two (400-350 ka, 280-250 ka) (Forester et al. in prep.). Levels of precipitation also differed from one glacial period to the next and were probably higher during the earlier, warmer glacial periods than during at least the last, cold glacial period. Interglacial periods also varied in temperature and precipitation (Carter 1993, 1997; Smith et al. 1997). Modern arid climates of the Holocene began about 8500 years ago and were the driest, and perhaps warmest, of all past interglacials (Forester et al. in prep.). Future climate is expected to begin moving towards glacial conditions within the next thousand years, assuming the cyclic characteristics of long-term (millennial) climate change based on the relationship between earth's orbital parameters and insolation, persist (Forester et al. in prep.). For additional information see Adkins et al. (1997), Broeker (1997), Winograd et al. (1997), Keigwin (1996), and DeWispelare et al. (1993).

To provide insights into the magnitude and variability of temperature and precipitation during the last glacial period, quantitative analysis of the packrat-midden (Neotoma sp.) database was performed, focusing on the time intervals 35 to 30 ka, 27 to 23 ka, 20.5 to 18 ka, and 14 to 11.5 ka. The results suggest considerable climatic similarity between the intervals (R. S. Thompson and K. H. Anderson, written communication; report in prep.). Further, when the results were evaluated relative to elevation, both the mean annual temperature and mean annual precipitation anomalies were found to be greater at lower than at higher elevations. The merged values suggest that the past temperature was about 9°C colder than present day at an elevation of 800 m, but only about 3°C colder at an elevation of 1,800 m. Precipitation rates were about three times the rates of present day at an elevation of 800 m, and 1.5 times the rates of present day at an elevation of 1,800 m. This pattern, in past climate parameters, is consistent with a postulated regional resident polar air mass that would have had a greater impact on the climate parameters of lower elevations than on higher elevations (R. S. Thompson and K. H. Anderson, written communication; report in prep.).

Studies of past discharge deposits continued to provide information for reconstruction of past climatic and hydrologic conditions at Yucca Mountain, and for development of possible future climatic scenarios. Analysis of deposits down gradient from the potential repository, and in the central Amargosa Valley, indicates that the fine-grained, carbonate-bearing deposits are associated with the past discharge of regional, saturated-zone groundwater (Paces et al. 1997c). New geochronological and isotopic analyses support the interpretation that groundwater from varied sources discharged during each of the last two glacial cycles (170 to 130 ka and 40 to 12 ka). Regional saturated-zone groundwater responded to climate-induced changes in effective moisture, infiltration, and recharge. For groundwater discharge to have occurred, the regional water table must have been elevated as much as 100 m (+20 m) above modern levels. Spring activity was synchronous at widely distributed sites. The most recent activity ceased around 12,000 to 16,000 years ago at higher elevations on the north side of the Amargosa Desert and slightly later (9,000 to 10,000 years ago) at lower elevations on the south side, consistent with the regional groundwater gradient. Isotopic data indicate that the discharge occurred from the regional carbonate aquifer rather than from the surface or from perched sources (Paces et al. 1996). Past discharge records were consistent with other regional records (Paces et al. 1997c) that indicate a greater mean annual precipitation and a lower mean annual temperature than exist at present. Because there was no evidence of substantial lag times between the onset of cooler, wetter pluvial conditions and high stands of the water table, it appears that the water table responds to changes in recharge rates within several millennia. A tectonic model of decline in water table elevation (Winograd et al. 1985; Winograd and Szabo 1988) was not supported by these studies (Paces et al. 1997c).

Geochronological studies of alluvial sediments in Fortymile Wash documented episodes of aggradation and incision of the fan surface, which are linked to variations in precipitation and stream flow associated with previous climates that existed at Yucca Mountain (Paces et al. 1997c). Aggradation occurs during warmer, dryer interglacial episodes, whereas incision predominates during colder, wetter climates. Incision during the wetter episodes indicates that Fortymile Wash sustained a seasonal, and possibly a perennial, flow capable of supplying recharge along its channel. A localized recharge in Fortymile Wash during pluvial periods could have significantly affected the underlying water table and could have changed the potentiometric surface from generally east-dipping to west-dipping in the vicinity of Yucca Mountain (Paces et al. 1997c).

Climate Simulations – The hydrologic effects of two sets of climatic conditions were simulated (D'Agnese et al. in prep.) using the existing regional groundwater flow model (D'Agnese et al. 1997). One simulation represented climatic conditions for 21,000 years ago when glaciation was at a maximum. The other simulation represented a possible future climatic condition when atmospheric carbon-dioxide concentrations might be doubled. For the 21,000-years ago conditions, the simulated groundwater recharge over the region was about five times the recharge rate of the present day. Under these climatic conditions, simulated water levels beneath the potential repository block were 60 m higher than present-day levels; water levels north of the large hydraulic gradient were up to 150 m higher than present-day levels; and existing hydraulic gradients in the region were enhanced. Under the conditions of doubled atmospheric carbon-dioxide, simulated recharge both increased and decreased relative to present-day rates, but overall recharge was approximately 1.5 times the present-day rate. The configuration of the potentiometric surface changed only slightly as several playa lakes in the northern and northeastern areas of the model domain became points of groundwater discharge. Simulated water levels beneath Yucca Mountain were less than 50 m higher than present-day levels, and existing hydraulic gradients in the region were enhanced.

To support the process model abstraction for the TSPA-VA, changes in the groundwater flow paths and flux rates for the two sets of climatic conditions were also simulated using the regional groundwater flow model. The southeastwardly groundwater flow paths from Yucca Mountain did not change significantly from the present-day flow paths for the two sets of climatic conditions. However, relative to the present-day flux rate, simulated groundwater flux beneath the potential repository area increased by approximately four times for the conditions 21,000 years ago and by about 1.5 times for the doubled carbon-dioxide conditions.

During FY 1998, the following activities will occur:

- Ongoing meteorological studies will provide weather and air quality data for compliance with State Air Permits and to support development of the environmental impact statement. Estimates of flood potential, and means and extremes of weather variables will be developed to support engineering design for surface facilities. Dispersion modeling will support development of the environmental impact statement and provide results for preclosure radiation safety analyses.

- A high-resolution paleoclimate data set from Owens Lake will be analyzed to establish probable bounds on climatic parameters for the next 100,000 years and for past "super-pluvial" conditions as input to performance assessment models. An estimation of bounds on future climatic conditions will involve developing a high-resolution paleoclimate data set from the Owens Lake sediment record for 400,000 years ago to 300,000 years ago (the period having earth orbital parameters most like those that are likely for the next 100,000 years). Development of the high-resolution data set will improve the understanding of climatic characteristics during the period 400,000-years ago to 300,000-years ago. The final results of the packrat midden (*Neotoma sp.*) analysis will be included in the regional climate maps showing the mean annual temperature and precipitation.

The evaluation of past discharge rates will focus on establishing first-order estimates of the flow rates of past groundwater discharge, by examining modern discharge at Ash Meadows and the recent deposits associated with this discharge. Because the flow rates of many of the springs in the area have been monitored historically, it should be possible to establish relationships between modern flow-rates and the lateral extents of associated discharge deposits. These relationships will then be scaled up to match the extent of past discharge deposits at Ash Meadows. The resulting flow rate estimates will be compared to those calculated by regional model simulations for the pluvial periods. If successful, these relationships will be used for other Amargosa Valley deposits to estimate past discharge rates for the Yucca Mountain flow system. Quantification of past groundwater discharge rates is expected to help constrain the saturated-zone flow models that are used to simulate either past or possible future hydrologic conditions that are wetter than present-day conditions.

5.8 DISRUPTIVE CONDITIONS

Disruptive conditions include natural hazards (such as earthquakes, volcanic activity and floods) and human intrusion (such as that associated with development and exploitation of natural resources), which have the potential to compromise the performance of the repository. Studies of disruptive conditions discussed in this section are generally limited to descriptions of the hazards associated with various disruptive conditions. For discussions of potential consequences to repository performance, see Chapter 3.

Natural Hazards

Although the occurrences of natural hazards are rare and impacts are generally believed to be quite small, the potentially disruptive effects of earthquakes and volcanism are being evaluated. They could affect the development of repository design and may have some effect on preclosure and postclosure repository performance. For the preclosure period, the repository design must consider expected vibratory ground motion, fault displacement, and ashfall. The postclosure performance assessment must address the tectonic effects that are both direct (vibratory ground motion, fault displacement, volcanic eruption through the repository) and indirect (seismic induced rockfall, seismic effects on the hydrologic system, volcanic effects on the hydrologic and geochemical systems).

During this reporting period, work continued on the probabilistic seismic hazard assessment. Three workshops were held. These workshops served as forums to provide experts on seismic source, fault displacement, and ground motion, with feedback on their preliminary evaluations. The proceedings of these workshops are summarized in three reports (CRWMS M&O 1997cn, CRWMS M&O 1997co, and CRWMS M&O 1997cp). Following these workshops, the experts finalized their interpretations to provide the basis for preliminary calculations of ground motion and fault displacement hazard.

The preliminary interpretations, along with a description of the processes used to obtain them, are documented in two additional reports (CRWMS M&O 1997cq and CRWMS M&O 1997cr). These describe seismic source, fault displacement, and ground motion characterization.

All the expert teams characterizing seismic sources defined volumetric zones to account for background earthquakes that are not explicitly included in their models of local and regional faults. Recurrence rates for these zones were derived from the catalog of historical seismicity. Regional fault sources were defined by most teams as faults within 100 km of Yucca Mountain that were judged to be capable of generating earthquakes with moment magnitudes of 5 or greater. This evaluation was based primarily on fault length and the histories of multiple surface fault rupture events during the Quaternary Period. Recurrence rates for regional fault sources were estimated either from fault slip rates or fault recurrence interval data. Maximum magnitudes were determined using regression relations between magnitude and various fault parameters such as length, rupture area, and maximum and average displacement. In characterizing local fault sources, the expert teams considered various fault behavioral and structural models. Most teams preferred a planar block model with linkages along strikes or down-dips. The simultaneous rupture of multiple faults was included by all teams as a possible model. As with the regional fault sources, a slip rate and a recurrence interval approach were used to assess recurrence rates for local sources, and a maximum magnitude was determined using regression relations between magnitude and various fault parameters. In addition to the volumetric zones, and regional and local faults, some teams also considered sources such as a regional buried strike-slip fault or volcanic sources. Teams that did not explicitly include these types of sources in their overall model considered them within their volumetric non-structure specific background source zones.

The fault displacement characterization involved evaluation of both principal and distributed faulting. Principal fault displacement is faulting along the main plane (or planes) of crustal weakness and is responsible for the release of seismic energy during an earthquake. Distributed faulting occurs on other nearby faults or fractures in response to the principal displacement. The expert teams used two approaches to determine the rate of fault displacement. In one approach, they used feature-specific or point-specific displacement data (recurrence interval and slip rate). In the other approach, the rate of displacement was derived from the earthquake recurrence rate for a particular fault. For the rate of distributed faulting based on earthquake recurrence, teams used empirical data on the spatial density of distributed faulting from historical events and considered the tendency of faults or fractures to slip based on their orientation relative to the stress field. To characterize the amount of displacement for principal and distributed faulting, teams used approaches based on maximum displacement per event, average displacement per event, or cumulative slip. The teams also assessed the uncertainty in the amount of displacement based on various analyses of available data.

For ground motion characterization, experts evaluated available empirical data and calculations of ground motion from numerical models to provide point estimates for a suite of magnitude, distance, and frequency combinations. In addition to median values, the experts' interpretations included ground motion variability and the scientific uncertainty. The suite of point estimates included earthquakes with a range of moment magnitudes from 5.0 to 8.0, distances from 1 to 160 km, normal and strike-slip faulting, and, for normal faulting, the differences in hangingwall and footwall motions. In addition, the experts provided adjustment factors for two special cases: parallel multiple faults and deep, shallow-dipping detachment surfaces. The ground motions developed were intended to characterize surface shaking at a hypothetical site with properties the same as those encountered at a depth of 300 m at Yucca Mountain. Preliminary seismic source, fault displacement, and ground motion characterizations were used to calculate preliminary ground motion hazard estimates (CRWMS M&O 1997c). For mean annual probabilities of exceedance of 10^{-3} and 10^{-4}, corresponding to Frequency-Category-1 and Frequency-Category-2 seismic design basis ground motion, the resulting peak horizontal ground accelerations were approximately 0.18 g and 0.55 g respectively, for the hypothetical reference site.

As part of the effort to characterize seismic hazards at Yucca Mountain, ground motions from NRC-defined Type I faults within 5 km of the conceptual controlled area boundary were also evaluated deterministically. During the reporting period, potential Type-I faults were identified, and their maximum magnitudes and distances from the site were characterized. The maximum magnitude was assessed on the basis of regression relations between magnitude and various fault parameters such as length, rupture area, and displacement.

Near-surface ground motion attenuation was determined for a suite of about 250 small earthquakes. The values obtained ranged from approximately 22 to 56 milliseconds, and were about 50 to 100 percent greater than estimates obtained using larger earthquakes. The difference in the results obtained from the smaller versus the larger earthquakes may be due to an underestimation of the radiated high frequency energy by the source model that was assumed for the larger events. These new results will be considered in developing seismic design inputs for a geologic repository at Yucca Mountain.

Human Intrusion

Disruptive conditions also include the likelihood that human intrusion into a repository will occur. One factor that could lead to human intrusion is the presence of natural resources at Yucca Mountain. Thus, site characterization studies assessed the potential for natural resources in the vicinity of Yucca Mountain. Assessments of natural resources completed during the reporting period addressed metallic minerals, hydrocarbons, water resources, and potential future water use.

Metallic Minerals

Analysis of extensive geological, geochemical, and mineralogical data (Castor et al. in prep.) indicates that the potential for metallic mineral deposits or naturally occurring uranium resources that could be exploited now or in the foreseeable future is low to nonexistent at Yucca Mountain. A simple statistical analysis indicates substantial differences in geochemical patterns for precious metals, base metals, and pathfinder elements typical of metal mining districts and those found at Yucca Mountain.

The geochemical patterns found at Yucca Mountain are not those that would be expected if deposits of metallic minerals were present. A few slightly elevated metallic geochemical values at Yucca Mountain were found in scattered occurrences of mineralized rock and alteration that are considered to be of fumarolic origin. These occurrences are thought to be of minimal size and have little potential for depth extension. The highest metallic value was from one sample of float (unattached surface rock) containing about 3 percent tin (representing about 62 pounds of tin per short ton) collected from one occurrence in the southern part of Yucca Mountain. Further examination of the occurrence suggested that similar rock samples from the same area contain less than 0.3 percent tin, and that only minor amounts of similar rock were present (Castor et al. in prep.). (Note: Based on U.S. Bureau of Mines data [Bleiwas et al. 1986], minimum grades of new lode tin (tin deposit on consolidated rock) resources would have to be at least 0.33 percent tin to be exploitable under current economic conditions). Small tin occurrences of the type found in southern Yucca Mountain probably are of fumarolic origin. Such deposits have never been mined in the United States (Castor et al. in prep.).

Hydrocarbons

Some of the basic elements of a viable petroleum system appear to be present in the Yucca Mountain area. However, known variations in conditions of potential source, reservoir, trap, and seal rocks at Yucca Mountain have negative implications for the accumulation of the hydrocarbons at the repository site (French in prep.). Based on comparison of the Yucca Mountain area to known producing fields in the Great Basin, the volume of potential source rocks and the hydrocarbon generation potential are limited, and potential seals are apparently missing at Yucca Mountain.

No evidence has been found for the accumulation of tar sand deposits in the region (Grow et al. 1994), and it is extremely unlikely that tar sands are concealed at depth below Yucca Mountain but not exposed in rocks of the surrounding area (Castor et al. in prep.). In the unlikely event that tar sands were present at Yucca Mountain, they would most likely be found in the Paleozoic marine rocks, which occur at depths of 1,200 m or more. Conventional recovery methods for tar sands require surface mining and processing of large volumes of rock. At such depths, stripping ratios would preclude surface mining, and mass mining underground at such depths is not economically feasible now or in the foreseeable future.

Coal

There are no reports of coal from southern Nevada in the vicinity of Yucca Mountain (Castor et al. in prep.). In addition, no coal or coal-bearing Tertiary sedimentary rocks have been encountered in drill holes on and adjacent to Yucca Mountain (Castor et al. in prep.). Because coal was actively sought during early mining and mineral exploration, it is unlikely that any coal beds of significance remain undiscovered. Therefore, potential for the existence of coal resources at Yucca Mountain is considered very low.

Water Resources

The Yucca Mountain area and the downgradient Amargosa Valley are within the Death Valley Hydrographic Region and within the Furnace Creek-Alkali Flat Groundwater Basin of that larger unit. A study by Woodward-Clyde (in prep.) suggests that current recharge to the Furnace Creek-Alkali Flat Groundwater Basin is about in equilibrium with discharge from the basin. This discharge is chiefly in the form of evapotranspiration by native vegetation in the vicinity of Ash Meadows and by irrigation in the Amargosa Valley. The average long-term annual recharge to the basin, termed the perennial yield, is estimated to be about 25,000 acre-feet per year.

Analyses of future water demand in the groundwater basin that includes Yucca Mountain by Nye County and the Project (Woodward-Clyde in prep.) indicate that future water use by the potential repository project would not be significant compared to demand for water for irrigation and for urban development in the Amargosa Valley. Regardless of the end use of the water, the long-term supply is inflexible and is limited to the perennial yield, which can only be increased by importation of water from other basins, or by changes in climate. Over time periods from centuries to millennia, significant increases in water demand from the groundwater basin, which includes Yucca Mountain, could not be supported because of the physical limitation of the natural supply.

In summary, the potential for human intrusion, for the purpose of exploiting and developing natural resources at Yucca Mountain, appears to be low for all resource types evaluated.

During FY 1998, the following activities will be completed:

- The probabilistic seismic hazard assessment will be completed and seismic design inputs will be developed. For the hazard assessment, the fault displacement hazard will be computed based on preliminary fault displacement models. A final feedback cycle with the experts will then be conducted based on the preliminary hazard results for ground motion and fault displacement. Seismic source, fault displacement, and ground motion characterizations will be finalized, and the final hazard calculations will be completed. A final report describing the probabilistic seismic hazard assessment is scheduled for completion in February 1998.

- For the deterministic assessment of ground motion from Type I faults within 5 km of the conceptual controlled area boundary, a ground motion spectrum will be obtained for each fault using the maximum magnitudes and distances assessed in FY 1997. The results of the study will be described in a report, scheduled for completion in December 1997.

- Design earthquakes will be identified. The design earthquakes will feature appropriate annual probabilities of exceedance, from the deaggregated probabilistic hazard results for a hypothetical rock outcrop with properties similar to those found at a depth of 300 m within Yucca Mountain. Based on these events, ground motion will be developed for a rock outcrop at the surface of Yucca Mountain, and for an interface at a depth of 300 m. Ground motion inputs will include peak acceleration and velocity, response spectra, time histories, and strain as a function of depth. Both vertical and horizontal components will be described. For fault displacement, the amount and direction of fault displacement will be taken directly from the hazard results. The methodology to develop seismic design inputs and the results of implementing the methodology will be presented in a report that is scheduled for completion in February 1998.

CHAPTER 6 - REFERENCES

The references cited in the text are available for inspection at the DOE public reading room located at 1180 Town Center Drive, Las Vegas, Nevada, in open literature, or through proceedings volumes for symposia and technical conferences. Technical reports and research products published by participating organizations on the Project may be obtained through the DOE Office of Scientific and Technical Information at Oak Ridge, Tennessee, which is the national center for dissemination of nonclassified scientific and technical information prepared from research sponsored by DOE, or the National Technical Information Service in Springfield, Virginia. These documents may be ordered from:

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Abstracts of Project-sponsored reports can be found in the Project Bibliography, which is updated approximately every six months, and may be obtained through the Office of Scientific and Technical Information.

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