SITE CHARACTERIZATION PROGRESS REPORT:
YUCCA MOUNTAIN, NEVADA

October 1, 1996 to March 31, 1997

Number 16

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YUCCA
MOUNTAIN
PROJECT

EXECUTIVE SUMMARY

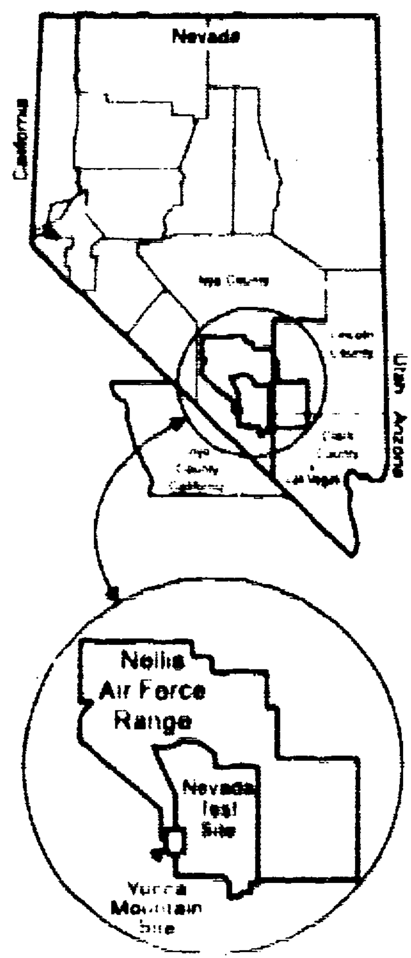
Site characterization activities at Yucca Mountain, Nevada, from October 1, 1996, through March 31, 1997. Sixteenth in a series reported in accordance with the requirements of Section 113(b)(3) of the Nuclear Waste Policy Act of 1982, as amended, and 10 CFR 60.18(g)

During the first half of fiscal year 1997, activities at the Yucca Mountain Site Characterization Project (Project) were focused on implementing the objectives of the 1996 Revision 1 Program Plan (revised Program Plan) issued by the Office of Civilian Radioactive Waste Management of the U.S. Department of Energy (Department).

To help support Project progress toward Plan objectives, the Department began re-evaluating its waste containment and isolation strategy in light of the new information about the percolation flux rates described in the last progress report. A revised summary of the strategy should be available next period. In addition, the Department published a Notice of Proposed Rulemaking to amend the repository siting guidelines and began taking public comments.

The fiscal year 1997 Energy Water and Development Appropriations Act provides that the Department shall complete a viability assessment by September 30, 1998. Scientific and engineering activities this period were directed toward providing information for the 1998 viability assessment. The Project adopted several measures to ensure the total system performance assessment to support the viability assessment would provide a complete and unambiguous record of the evaluation for traceability and be readily understood by the reader and reviewer (transparency). A major effort toward those ends was the conduct of seven abstraction-testing workshops designed to ensure that the performance assessment properly reflects the comprehensive process models for natural and engineered systems developed by principal investigators. In addition, formal expert elicitations began on two of the important foundational process-level models for the potential repository system at Yucca Mountain — the site-scale unsaturated zone flow model and the waste package degradation model.

Design activities continued to focus on resolving key design issues and on designing systems, structures, and components of the repository system that have little or no regulatory precedent and have a major impact on performance assessment, schedule, construction, and cost.



Location of Yucca Mountain site

Two tests in the thermal testing program are under way and a third is being planned and constructed. These tests will provide information on how the heat generated by the emplaced waste will affect the rock and the fluids in the repository system. The single-heater test began in August 1996, and preliminary results are becoming available. The large block test began in February 1997. Test setup is under way for the drift-scale heater test, scheduled to begin in December 1997.

The following sections summarize progress toward achieving Program Plan objectives and report the Project's technical progress.

OFFICE OF CIVILIAN RADIOACTIVE WASTE MANAGEMENT PROGRAM

The Department's revised Program Plan, released last reporting period, outlines a focused, integrated program of site characterization, design, engineering, environmental, and performance assessment activities that will achieve key policy and statutory objectives. The revised plan modified the 1994 Program Plan to reflect the increased understanding of the Yucca Mountain site and the ongoing policy process.

The revised plan identifies three near-term objectives for the Project: (1) updating in 1997 the regulatory framework for repository siting to be consistent with a more focused program driven by the results of performance assessment; (2) supporting a 1998 viability assessment of Yucca Mountain; and (3) if the site is suitable, submitting a Secretarial site recommendation to the President in 2001, and a license application to the U.S. Nuclear Regulatory Commission (Commission) in 2002. Outlined in the following sections are the bases for the near-term objectives, the Department's strategies for achieving those objectives, Project progress in achieving those objectives, and the Department's waste containment and isolation strategy. Details on Project progress are presented following this Program summary.

Regulatory Framework Update

To achieve the first near-term objective, the Department is reviewing and proposing to revise its siting guidelines (10 CFR Part 960). The guidelines were first promulgated when multiple sites were under consideration. Over the past decade, however, legislation has dictated characterizing a single site and developing site-specific standards to protect the public. During this time period, the technical understanding of the site has increased dramatically, especially after the Exploratory Studies Facility provided underground access. Thus, the Department proposed to update its siting guidelines by adding a subpart that would focus specifically on the overall preclosure and postclosure performance of the potential repository system at Yucca Mountain.

The Department's proposed amendments to the guidelines would refer to the site-specific radiological protection standard being developed by the U.S. Environmental Protection Agency (Agency) pursuant to the Energy Policy Act of 1992. The Agency's site-specific standard will appear in 40 CFR Part 197. The Energy Policy Act also requires the Commission to implement the Agency standard by revising the disposal regulations (10 CFR Part 60) within one year after the Agency Administrator promulgates the site-specific standard to make the regulations consistent with the standard. In the interagency review process, the Department will provide input to both the Agency and the Commission regarding their revisions to help ensure the resulting regulatory framework can be implemented. Since the last progress report, however, neither the status of the standard nor the Department's view on issues relevant to implementing the standard has changed.

In modifying its siting guidelines, the Department is proceeding in accordance with a public rulemaking process that began December 16, 1996, with the publication of a Notice of Proposed Rulemaking. In addition to accepting written comments, in January the Department held public hearings in Las Vegas, Nevada. The public comment period is scheduled to close next reporting period, and the Department expects to issue a final rule in mid-late 1998. If implemented, the regulatory changes would streamline the process for site evaluation and repository development, while protecting public health and safety and the environment.

The Department published its Notice of Proposed Rulemaking on December 16, 1996, beginning a public process to update the siting guidelines (10 CFR Part 960)

Viability Assessment

During this reporting period, Project activities have concentrated on supporting the upcoming viability assessment. The viability assessment is not the same as the site suitability determination and site recommendation. Rather, it is an interim step toward that recommendation. The viability assessment will serve two purposes: first, to guide the completion of the work required for an evaluation of site suitability and preparation of a license application, and second, to provide policy makers with a better estimate of the viability of a geologic repository at the Yucca Mountain site. The viability assessment is a logical convergence point at which the Department can make an improved appraisal of the prospects for geologic disposal at Yucca Mountain using the results of the program first described in the 1988 Site Characterization Plan and the results from excavation of the Exploratory Studies Facility. These results allow the Department to develop an integrated picture of the repository system at the site that was previously impossible. The information the Department produces for the viability assessment will be an important validation of the practicality of the existing national policy of geologic disposal, and will allow policy makers to make a measurably improved judgment of the prospects of recommending Yucca Mountain for repository development and licensing to authorize construction.

The viability assessment will include the following:

1. The preliminary design concept for the critical elements for the repository and waste package
2. 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
3. A plan and cost estimate for the remaining work required to complete a license application
4. An estimate of the costs to construct and operate the repository in accordance with the design concept.

The Department has developed plans for all four components of the viability assessment, as discussed in the following sections.

Current repository and waste package design activities are focusing on design elements critical to waste isolation and radiological safety that have little or no regulatory precedent.

Repository and Waste Package Design. 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 and have a major impact on performance assessment, schedule, construction, and cost. When complete, the viability assessment design effort will have evaluated the technological feasibility of the conceptual designs but will not have developed all the detail needed for licensing.

During this reporting period, the Project continued to modify designs to reflect the de-emphasis on the multi-purpose canister and to accommodate receipt of mainly uncanistered spent nuclear fuel. Sensitivity studies for the higher percolation fluxes at the repository level that were reported last period were performed on the engineered barrier system. Laboratory tests on waste package materials and waste forms continued to provide input to degradation process models. In addition, both laboratory and field tests are being conducted to assess changes in concrete caused by the hydrothermal cycle. These tests will be used to support a design decision on using concrete for mechanical support in the repository emplacement drifts. The Commission's revision of 10 CFR Part 60 on design basis events became effective this period, and the Project grouped and prioritized analysis events previously identified for consideration as design basis events. This rule change resulted in limits and models for Project design basis events that are more consistent with those of the commercial nuclear industry.

Total System Performance Assessment. Total system performance assessment, the second component of the viability assessment, is a key element in the Department's performance-driven program. This assessment will evaluate the range of probable behavior of the repository in the Yucca Mountain geologic setting.

To provide a valid and defensible evaluation of the site's performance, the Project is developing an integrated total system performance assessment to support the viability assessment. Important concerns are that model development be focused on issues that are most important to performance and that the evaluation ensure traceability (provide a complete and unambiguous record) and transparency (be easily understood by the reader and reviewer).

Measures to Ensure Traceability in Total System Performance Assessment

- *Abstraction-Testing Workshops*
- *Formal Expert Elicitation*
- *Peer Review*

To ensure traceability, the Project has adopted three processes along with accompanying records: a series of abstraction-testing workshops, a formal expert elicitation process, and peer review.

The first process, the abstraction-testing process, is used to ensure that the total system performance assessment properly reflects results from the highly detailed and computationally intensive site and engineered system process models. The models characterize processes and features of both the natural and the engineered systems and represent the work of site characterization, design, environmental programs, and performance assessment. In the probabilistic total system performance assessment calculation, abstracted models are used as surrogates for the comprehensive process models. The abstracted models must, however, maintain the essential elements of the process models, including key interdependencies. This abstraction-testing process is critical to the success of the Project.

This period, Project staff prepared for, conducted, and evaluated the results of several abstraction-testing workshops. During the seven workshops held this period, Project staff developed a list of criteria against which all issues were then ranked for importance to postclosure performance. From this ranking, the highest priority issues were identified. Finally, a short synopsis for the abstraction-testing plans was developed. Following the workshops, work began on completing the details of the plans that will guide the implementation of the analyses identified in the workshops. The results will be the parameters, process models, and alternative concepts used in the total system performance assessment supporting the viability assessment. Two more abstraction-testing workshops are scheduled for next period.

The second process to achieve traceability is a formal expert elicitation, which is used to characterize the state of existing knowledge in an area. The expert elicitations follow the guidance provided in the Commission's Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program. The expert elicitations are used (a) to quantify and document the uncertainties in the process models to strengthen the assessment and (b) to involve outside experts in the evaluation of all available data. Formal expert elicitation is currently under way on the site-scale unsaturated zone flow model—one of the most important of the foundational process-level models.

For peer review, the third process, an external panel of experts has been established to monitor and review the preparations for the assessment, as well as the final product. The peer review panel will first review the previous total system performance assessments and then make observations on the plans, approach, and assumptions for the performance assessment to support the viability assessment. The reviewers will also review the process modeling and the performance assessment abstraction process. Finally, they will provide a formal peer review of the assessment supporting the viability assessment, and the comments and recommendations will be incorporated into the performance assessment to support the license application. The peer review panel convened this period, and member orientation began in February 1997.

Measures to Ensure Transparency in Total System Performance Assessment

- *Graphic presentation*
- *Electronic hypertext*
- *Computerized data retrieval and selection system*

Besides the three steps to ensure traceability, the Project is also using specific measures to ensure transparency. There is an initiative under way to examine ways of graphically presenting the total system performance assessment results to make them more easily understood by those who are not performance assessment specialists. The Project is also investigating using hypertext to increase the reviewer's electronic

Site Recommendation and License Application

After the viability assessment, the Project will prepare the additional information required for site recommendation and the license application. If the site is found suitable in accordance with the then-existing siting guidelines, the Secretary of Energy will issue a site recommendation in 2001, following public hearings to be held in the State of Nevada before a possible site recommendation. If the site is approved by the President and permitted to take effect after submission to the U.S. Congress, a license application will be submitted in 2002 requesting authorization to construct a repository. This schedule will allow repository operations to emplace waste beginning in 2010.

During this reporting period, the Project resumed preparation of the Environmental Impact Statement that would accompany any site recommendation. Preparation was deferred earlier after the public scoping period because of budget constraints. Work also began on several documents that will be key to developing the license application: the Project Integrated Safety Assessment, the Technical Guidance Document for License Application Preparation, and the License Application Management Plan. The Project Integrated Safety Assessment will summarize the current knowledge about the site obtained from scientific investigations, design, and performance assessment. The document will become a starting point for the license application by updating and using the information in the safety analysis report that supports the license application. The technical guidance document will provide authors with detailed content guidance, regulatory requirements, and acceptance criteria for the license application. The License Application Management Plan establishes the process for managing the development of the license application.

The objectives of Department and Commission interactions are reaching a common understanding of issues significant to overall repository performance and agreement on methods and approaches to important technical issues.

Interactions between the Department and the Commission are an important part of the prelicensing period before a decision is made whether to recommend the site. The Department believes that interactions with the Commission staff should focus on two objectives: (1) reaching a common understanding regarding the issues that are significant to overall repository performance, and (2) reaching agreement on the adequacy of proposed methodologies and approaches to address important technical issues, such as criticality control and seismic design. The goal of Department and Commission interactions is to reach a mutual understanding of the repository concept as it develops. This understanding will provide a basis for the Commission's preliminary comments (to be included in the Department's site suitability package) on the sufficiency of site characterization and design for inclusion in a license application. During this reporting period, the Department continued interactions with the Commission staff to help resolve the issues of seismic hazards, igneous activity, and repository criticality. Many of the interactions involved resolving Commission comments on the Department's methodology reports.

Waste Containment and Isolation Strategy

The Department's waste containment and isolation strategy is being used to focus the efforts leading to the viability assessment and to guide the work beyond the viability assessment. The two technical objectives of the strategy are (1) to limit the annual dose to members of the general public following permanent closure of the repository and (2) to provide total containment of the waste within the emplaced waste packages for several thousand years during the period of highest radionuclide inventory and temperature. The strategy outlines the Department's approach to addressing and resolving postclosure performance issues and also focuses the science and design work needed to determine postclosure performance.

Percolation flux values affect all the system attributes identified in the waste containment and isolation strategy as most important for predicting the performance of the engineered and natural barriers.

After a summary version of the strategy was completed and distributed in July 1996, new information derived from interpretations of several site investigations indicated the potential for average percolation flux values at the repository horizon in the range of 1 to 10 millimeters per year or potentially higher, with part of this flux associated with fast pathways. Percolation flux affects all the system attributes identified in the strategy as most important for predicting the performance of the engineered and natural barriers: rate of water seeping into the repository, waste package lifetime (containment), rate of release of radionuclides from breached waste packages, radionuclide transport through engineered and natural barriers, including dilution in the saturated zone below the repository. Higher percolation flux may result in an increase in flux into the repository and a reduction of the time radioactive particles take to travel between the repository and the accessible environment. Also, higher percolation flux is expected to increase the relative humidity in the waste emplacement drifts, potentially leading to shorter waste package lifetimes. One of the highest priorities of the Project continues to be reducing the uncertainty in the range of percolation flux that can be expected at the repository horizon. A revised summary version of the strategy is scheduled to be issued next reporting period.

Specific progress in site investigations, design, and performance assessment supporting the revised Program Plan is discussed in the following section on Project progress.

PROJECT PROGRESS

Project activities this reporting period supported the objectives of the revised Program Plan, particularly the viability assessment. The following sections report progress in site investigations, repository design, waste package, performance assessment, Exploratory Studies Facility design and construction, and Project programmatic activities. Specific details relating to Project activities are presented in the respective chapters of the report and in supporting technical papers and reports referenced throughout the main body of the report.

Site Investigations and Analyses

Site investigations and analyses this period supported the development of the viability assessment through continued surface-based testing, testing in the Exploratory Studies Facility, and laboratory testing. In addition, analyses and process model development continue to provide information and input for design and the total system performance assessment components of the viability assessment. The major technical questions focus on thermal effects, hydrologic properties of major faults, percolation flux, saturated zone characteristics, and implications of climate change for repository performance.

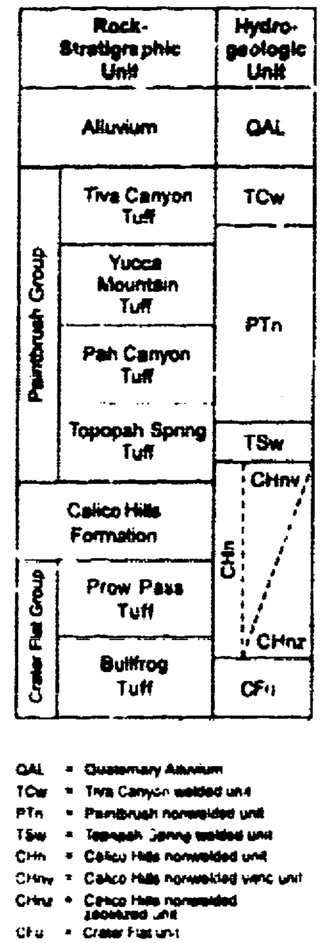
The Project continues to collect data to reduce the uncertainties in the range of percolation flux that can be expected in the repository horizon.

Unsaturated Zone Characterization. The Project continued its modeling, testing, analysis, and data collection activities to reduce the uncertainties in the range of percolation flux that can be expected in the repository horizon. Evidence on percolation flux continues to accumulate from a suite of ongoing testing in the Exploratory Studies Facility and surface boreholes that includes environmental isotope and fracture-coating studies, temperature monitoring, perched water evaluations, moisture monitoring, and pneumatic testing.

Systematic and feature samples in the ESF continue to show chlorine-36 from weapons testing in a few distinct fractured or faulted zones, indicating that at least a component of the water is less than 50 years old. Locations where multiple samples indicate chlorine-36 from weapons testing appear to be associated with major faults mapped at the surface. Rapid penetration of surface water to repository depth seems to occur only where faults cut all the way through the Paintbrush nonwelded hydrogeologic unit (*PTn* in the diagram) overlying the repository host rock. The amount and distribution of chlorine-36 from weapons testing found at the repository horizon seem to depend on the rate of surface water infiltration into and through the Tiva Canyon welded hydrogeologic unit (TCw). Distribution also seems to depend on the presence of small subsidiary faults or interconnected joints in the Topopah Spring Tuff (TSw) that cause the downward percolating water to spread laterally within the Topopah Spring Tuff away from fault zones such as the Sundance fault.

Tritium analysis of water extracted from rock samples in the Bow Ridge Fault Alcove of the Exploratory Studies Facility confirms the indications from chlorine-36 samples that water is transmitted rapidly through the Tiva Canyon welded hydrogeologic unit along the Bow Ridge fault. Likewise, tritium from weapons testing extracted from rock samples from the Upper Paintbrush Tuff Contact Alcove, which is located at the base of the Tiva Canyon welded hydrogeologic unit, suggest the relatively fast transport of water through the Tiva Canyon welded hydrogeologic unit above the potential repository horizon.

Other studies indicate that percolation flux rates may decrease with depth because of lateral flow both above and below the potential repository horizon. Estimates of unsaturated zone percolation flux from differences in temperature data between the saturated zone and the unsaturated zone in three boreholes indicate percolation flux rates of about 5 to 24 millimeters per year. But differences in temperature within the unsaturated zone alone (Topopah Spring Tuff and Calico Hills Formation) in two of the boreholes indicate lower rates of about 2 to 5 millimeters per year. Studies of the ratios of uranium-234 to uranium-238 in calcite and opal veins also point to decreases in percolation flux with depth. The ratios in samples from the Tiva Canyon Tuff, the Paintbrush nonwelded hydrogeologic unit, and the upper part of the Topopah Spring Tuff indicate larger rates of percolating water. The ratios in samples from the repository horizon, however, seem to indicate that relatively smaller rates of percolating water reach the repository horizon.

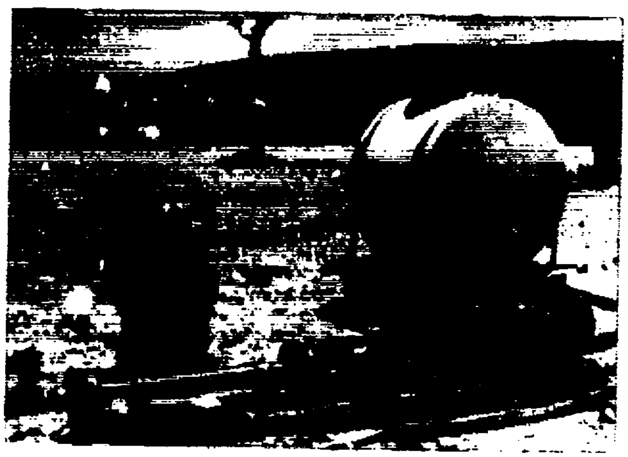


Correlation of hydrogeologic units with rock stratigraphic units.

Moisture monitoring in the Paintbrush nonwelded hydrogeologic unit in the south ramp of the Exploratory Studies Facility offered some clues about water potential and also ventilation effects. The initial measurements taken from instruments indicated that the water potential is considerably higher than indicated by measurements from boreholes. The data suggest a higher percolation flux through the Paintbrush

Nationally recognized experts are meeting to evaluate unsaturated zone flow characterization and modeling nonwelded hydrogeologic unit than would have been estimated using borehole data. Matrix flux in the Paintbrush nonwelded hydrogeologic unit has not yet been estimated using this new information.

Characterizing percolation flux in the unsaturated zone is critical to the Project's waste containment and isolation strategy, and the unsaturated zone flow model is therefore one of the most critical process-level models to be incorporated into the total system performance assessment supporting the viability assessment. Thus, the first expert elicitation to support the viability assessment has been started on the unsaturated zone model. A series of three workshops was conducted involving data collectors and analysts, modelers, seven nationally recognized experts in the field of unsaturated zone flow characterization, and observers. The purpose of the elicitation is to assess quantitatively the uncertainties associated with the model predictions of the spatial and temporal distribution of percolation flux.



C-hole testing

Saturated Zone Testing. Reactive and conservative tracer testing was completed in the Bullfrog-Upper Tram interval of the C-hole complex, the most productive zone of the volcanic aquifer. Tracer tests are used to estimate flow and transport parameters. These parameters are input to numerical flow and transport models and will be used in transport calculations to support the total system performance assessment for the viability assessment. Ultimately, the results will help predict likely dilution rates for radionuclides released from the repository, as well as the probable ground-water travel time for the radionuclides to move from the repository to the accessible environment.

Results from the C-hole pumping tests indicate that the system is a dual porosity flow and transport system. Tracers travel in the fractures but also diffuse into the rock matrix. This process is expected to increase travel time and enhance sorption. Matrix diffusion and sorption appear to be effective retardation and dilution mechanisms. The Project plans additional testing throughout 1997 on the low-flow zone (Prow Pass interval).

The studies indicate that even if a small fraction of the radionuclides reaches the water table quickly (within 10,000 years), the saturated zone would significantly dilute radionuclide concentrations before they reach the accessible environment. Thus, the saturated zone provides a defense against the most uncertain aspects of unsaturated zone performance; namely, that a fraction of the inventory could be rapidly transported to the water table.

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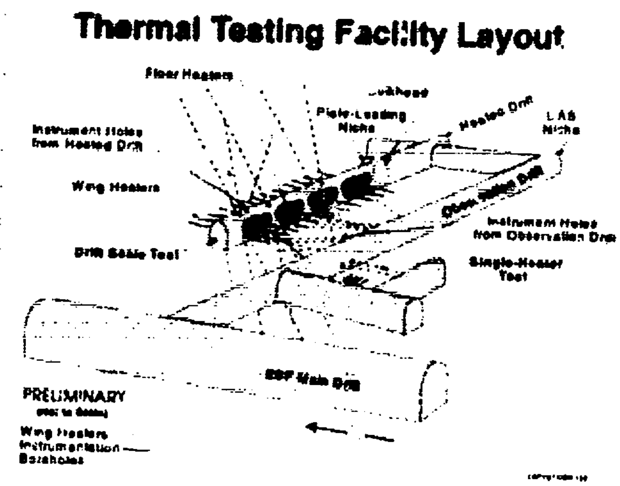
Transport. Transport activities this period continued to provide evidence about the potential radionuclide transport through the natural system. The migration (or transport) of radionuclides is affected by a variety of physical and chemical factors in the natural system, one of the most important being the amount and rate of flowing water.

A workshop was conducted to ensure proper abstraction and testing of the process-level model on unsaturated zone radionuclide transport. This workshop effort is part of the process to ensure that the total system performance assessment for viability assessment is valid and defensible. At the workshop, an abstraction-testing proposal was developed for sensitivity studies to examine the effects of mineral alteration on unsaturated zone radionuclide transport. Results of this workshop suggest that repository-induced alteration of existing minerals and glasses in Yucca Mountain tuffs could change the hydrologic properties (permeability and porosity) and geochemical properties (sorption capacity, composition including water) of the natural media over time. The end result of the reactions is that the unsaturated zone could be different in the future than it is today, and simulations of radionuclide transport in the unsaturated zone should consider possible changes in the rock properties. Abstractions will continue throughout the fiscal year.

Results of a workshop suggest that repository-induced alteration of existing minerals and glasses in Yucca Mountain tuffs could change the hydrologic properties (permeability and porosity) and geochemical properties (sorption capacity, composition including water) of the natural media over time. The end result of the reactions is that the unsaturated zone could be different in the future than it is today

Altered Zone. Modeling this period evaluated potential flow barriers in the altered zone—the region around the repository that may be changed by the heat of the emplaced radioactive waste. Reactive transport simulations in the altered zone indicate that when a flux of 100 millimeters per year is driven into the rock by the heat from the waste, solids could precipitate and seal fractures in less than 100 years. Such blockages would then force fluid to drain through the pillars between emplacement drifts. This water would thus bypass the waste without contacting it.

Thermal Testing. Three major in situ tests are being conducted to determine effects of heat from emplaced radioactive waste on surrounding rocks. The single-element heater testing that



Schematic of Thermal Testing Facility showing single heater test and drift

began last period continued in the Thermal Testing Facility. In this test, the heat comes from electric heaters that simulate the heat from emplaced radioactive waste in a repository. Some preliminary results indicate that convective heating effects are smaller than predicted from the thermohydrologic analyses. This may be because vapor escaping the block or heat loss through fracture systems is not accounted for in the modeling. Indications are that at temperatures above boiling, temperature predictions agree quite well with measured values. The heating phase of the single-heater test will be continued until the end of May 1997, when a decision will be made to continue the heating phase for another three months or to begin a cooling phase. Data will be taken throughout the heating and cool-down phase.

A test configuration has been developed for the drift-scale test scheduled to begin at the end of 1997. This test will be used to predict and measure coupled thermal-mechanical-hydrological-chemical processes. The test consists of a single heated drift 5 meters in diameter with electrical heater canisters placed in the drift to simulate waste packages. The test will provide information on temperature distribution and heat transfer modes; on the propagation of drying and re-wetting regions; on changes in water chemistry and mineralogy; and on thermal expansion and deformation modulus. Two to four years of heating are planned, depending on the processes observed in the first two years.

Large Block Test

The large block test to study coupled thermal-mechanical-hydrological-chemical processes began at Fran Ridge. The large block test is occurring in a medium with controlled thermal and moisture boundaries and with known multiple fractures and inhomogeneities. Instrumentation and waste package materials will be tested in a quasi in situ environment. Instrumentation was installed and the heaters turned on in February 1997. The heating phase of the large block test will continue throughout most of the next reporting period. The interior block temperature will be raised to approximately 140°C, with the temperature at the top of the block kept near 60°C, and these conditions will be held stable for about a month. After that time, the heaters will be turned off to start a cool-down phase. Data acquisition will be continuous during the heating and cool-down phases.

Geology. Mapping activities and data from alcove boreholes continue to provide a clearer picture of the faults affecting the Yucca Mountain central block. Mapping of the central block of Yucca Mountain has given geologists a better description of intrablock faults, such as Ghost Dance, Abandoned Wash, and Busted Butte. These faults spread out toward the surface into a series of branching faults from convergence at depth. The mapping indicates the pattern is maintained for block-bounding faults. This fundamental understanding of fault geometry will help geologists understand the deformation associated with Solitario Canyon fault, which bounds the western edge of the repository area.

Projection of the surface trace of the Ghost Dance fault to the location of the main trace indicates that the fault is nearly vertical.

Recent drilling in the Northern and Southern Ghost Dance Fault Alcoves has confirmed that the contact between the repository horizon and the upper lithophysal zone of the Topopah Spring Tuff is within 1 meter of the location predicted by the three-dimensional lithostratigraphic model. Video logs and core from the horizontal borehole in the Northern Ghost Dance Fault Alcove showed that the main trace of the Ghost Dance fault is about 154 meters east of the main drift of the Exploratory Studies Facility. Projection of the surface trace underground of the fault to the location of the main trace indicates that the fault is nearly vertical.

A study of potential magma sources in the Yucca Mountain region was completed. The results show that there is no large low-velocity zone under Crater Flat or Yucca Mountain that would suggest a major near-surface source of magma. This in turn corroborates some of the assumptions made in the Probabilistic Volcanic Hazard Analysis.

The process to develop ground motion and fault displacement information for design and performance assessment advanced this period with the resumption of the Probabilistic Seismic Hazards Assessment process. The assessment consists of two parts: (1) seismic source and fault displacement characterization and (2) ground motion characterization. The process uses panels of experts and formal elicitation of experts to examine data and interpretations. Several workshops and elicitation meetings were held this period. The final assessment will provide input to the final calculations of the annual probabilities of varying levels of ground motion and fault displacement. The process will be completed next reporting period.

Several branches of evidence are converging to provide sufficient understanding of past climate cycles to be able to generally forecast timing and magnitude of climate change in the Yucca Mountain area.

Climate. Paleoclimate records have been further refined, allowing a better understanding of past climate cycles in smaller time increments. Also, several branches of evidence are converging to provide sufficient understanding of past climate cycles to be able to generally forecast timing and magnitude of climatic change in the Yucca Mountain area. Analysis of diatoms (microscopic fresh water plants with silica shells) and ostracodes (small fresh water crustaceans) indicates extremely rapid changes in the climate at Owens Lake, California. The lake, which is a source of data on regional Quaternary climate, often varied from an overflowing, fresh water system to a closed, saline or even dry system in less than 1,000 years. These kinds of changes are more typical of interglacial and transitional periods than of the glacial periods. Ostracode records from about 55,000 years ago to present show very rapid shifts from dry climates similar to those of today to brief periods of warm, wet climates supported by summer rains.

Scientists have found that several types of records of past climate correspond well with each other. These include the isotope records at Devils Hole, Nevada, and the paleoclimate records from Owens Lake, California. In addition, both these records correspond to the marine oxygen isotope records that document global change in the earth's temperature and ice volume. The records indicate that the timing and rate of past climate change at Yucca Mountain coincided with the large, cyclic changes in global climate throughout the Quaternary. Using this knowledge and the knowledge that global climate shifts are related to changes in total solar heating, researchers think it may be possible in general terms to forecast the timing of climatic change in the Yucca Mountain area. Similarly, using the knowledge about how past magnitudes of climate change are related to particular segments of the solar radiation cycle, they think it may be possible to forecast the magnitude of future climate change in the region. Specifically, because the present-day segment of the solar cycle resembles that of about 400,000 years ago, the characteristics of the climate of that time may be expected to generally recur in southern Nevada.

Key Design Issues

- *Performance confirmation concept*
- *Engineered barrier system performance*
- *Waste handling capability*
- *Emplacement drift ground support concept*
- *Thermal loading*
- *Retrievability concept*
- *Remote control operations*
- *Disposal of site-generated waste*
- *Repository subsurface mapping strategy*
- *Postclosure performance standards*
- *Criticality control*
- *Repository seals*
- *Regional service agents/interim storage facility interface*
- *Additional waste forms*
- *Waste package sizes and weights*
- *Waste package materials*
- *Design basis model*
- *Subsurface development*
- *Surface development*
- *Site development*

Repository Design Activities

The Project continues its philosophy of a phased and evolving design that will support the viability assessment, environmental impact statement, site recommendation, and license application. Current work is focused on resolving key design issues and on designing systems, structures, and components that have little or no regulatory precedent and have a major impact on performance assessment, schedule, constructability, and cost. Even though the key design issues do not have to be completely resolved to support the viability assessment, sufficient progress is required to limit future redesign.

Regulatory Basis. In keeping with its design philosophy, the Project began developing technical guidance based on regulatory requirements to support its phased approach. The Project began identifying appropriate acceptance criteria for important systems, structures, and components in support of phase one design for the viability assessment. Later, the guidance to support phase two design (design development to support the license application) will be developed. The Engineering Compliance Plan will document the guidance for use by the design organization and the authors of the engineering chapters of the eventual license application. The plan will identify the information necessary to provide reasonable assurance to the Commission that the repository design supports construction of a repository that would not pose an unacceptable risk to the health and safety of the public or the repository workers. The plan will also identify regulatory guidance and industry standards that may apply to the repository design.

Regulatory guidance to support design will be documented in the Engineering Compliance Plan

The rule change to 10 CFR Part 60 incorporating design basis events became effective this period. In compliance with the requirements, the Project continued to refine its preliminary set of design basis events identified last period and also began analysis of the design basis events. Previously selected external events (caused by factors not directly related to repository design or operation) and internal events were grouped into analysis groups and the groups were prioritized on the basis of their potential impact on repository design, availability of information to support the analysis, and whether the analysis is needed to support the viability assessment.

Two pilot analyses were begun that will serve as templates for other consequence analyses that will be performed in future reporting periods. These pilot analyses are the first to incorporate revised radiological safety criteria from the revised 10 CFR Part 60. Analysis of design basis events will be used to refine the list of repository and waste package systems, structures, and components subject to quality assurance requirements and to help determine the level of design detail required to support the viability assessment and the license application.

Two pilot analyses were performed that are the first to incorporate revised radiological safety criteria from the revised 10 CFR Part 60

The repository design and emplacement concepts continued to be refined to minimize the amount of excavation necessary to emplace waste.

Supporting Activities. Supporting activities included improving the efficiency of the subsurface layout and analyzing activities that will impact repository operations.

The repository design and emplacement concepts continued to be refined to limit the amount of excavation necessary to emplace waste. Three potential space-saving concepts were identified. First, analysis of areal mass loading indicated that raising the loading to about 85 metric tons of uranium per acre would mean the repository area could be reduced slightly (2 percent) from that given in the advanced conceptual design while still meeting thermal goals. Second, minimizing drift space surrounding defense high-level waste could reduce the emplacement area required by about 10 percent. Finally, for a given areal mass loading, using a wider drift spacing coupled with closer waste package spacing in the drifts could also reduce the amount of excavation needed. The concepts have not yet been approved for implementation in the design, and further work to determine whether to implement them is in progress.

The Project analyzed the effect on Waste Handling Building operations of changing to receipt of predominantly uncanistered fuel

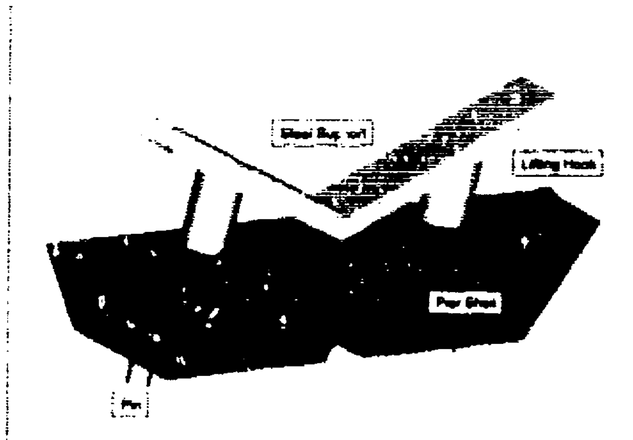
The Project analyzed the effect on Waste Handling Building operations of changing to receipt of predominantly uncanistered fuel. The analysis compared wet and dry handling concepts and preliminarily recommended a preferred waste handling approach that would include a staging area with five operation lines (three wet lines for uncanistered assemblies and two dry lines for canistered wastes). Handling operations for shipping casks and canisters, bare and canistered fuel, and disposal container operations were modeled. Also analyzed were staffing and shielding requirements. Analyses continue and the results will eventually be reflected in future revisions of the concept of operations and system design documents.

Waste Package Design Activities

Waste package design continued to emphasize the receipt and handling of uncanistered fuel because of the de-emphasis on the multi-purpose canister effort. Work this period focused on thermal, structural, and criticality analyses; and on the selecting of materials for waste packages support and inverts.

Design Analyses. Design analyses included thermal, structural, and criticality work. Thermal design efforts have advanced in three main areas: (1) evaluating the repository and emplacement drift thermal behavior and its impact upon waste packages, (2) evaluating waste package thermal conditions with regard to meeting the licensing requirements, and (3) evaluating and designing the waste package support and invert based on its thermal and structural performance under nominal repository conditions.

Two major structural design analyses are in process: (1) preliminary design of the waste package support and pier, and (2) waste package structural analyses. The objective of the first analysis is to determine appropriate dimensions and materials for the waste package support and pier on the basis of structural requirements. The purpose of the second is to determine component dimensions. Component dimensions are required to show the adequacy of the uncanistered fuel waste package design with stainless steel-boron neutron absorber plates under loading encountered during waste package drop events.



Waste Package Support and Pier Preliminary Design

The criticality activities consisted of developing inputs in support of Revision 1 of the Disposal Criticality Analysis Methodology Technical Report, developing the technical basis for integral principal isotope burnup credit, evaluating waste package designs for criticality control, and meeting with Commission staff to discuss Revision 0 of the methodology report. Revision 1 of the technical report is scheduled to be released late in fiscal year 1997. These activities are supporting the development of the Disposal Criticality Analysis Methodology Topical Report, scheduled to be completed and submitted to the Commission in 1998.

<u>Near-Field Studies</u>. Sensitivity studies related to potential percolation flux continued. One sensitivity study investigated the relationship between drift seepage and percolation flux for both homogeneous and heterogeneous rock conditions. The modeling showed that when heterogeneity in fracture properties increased, the threshold percolation flux at which water is able to seep into the drift decreased. In homogeneous conditions, narrowing the fracture aperture distribution also reduced the predicted threshold percolation flux.

A second sensitivity study investigated the influence of percolation flux on temperatures in the drift-scale test. The modeling indicated that for a 5 millimeter per year flux (a value chosen as representative of current predictions of percolation flux), the maximum drift-wall temperature at the center between the two ends of the heated drift would be more than 100°C lower than for a flux of 0.05 millimeters per year.

A third sensitivity study examined the sensitivity of fracture flow to percolation flux. Greater water pressure (as would be expected for higher percolation flux) was found to strongly increase fracture flow.

Both laboratory and in situ tests are being conducted to evaluate the hydrothermal alteration of concrete.

The Project is conducting both laboratory and in situ tests to evaluate the hydrothermal alteration of concrete. Experimental results will support a design decision about the use of concrete in the repository and a specific decision on the use of precast concrete liners for mechanical support in repository emplacement drifts. Laboratory tests are being used to determine microstructural, mineralogical, and mechanical changes in concrete and changes in water chemistry because of the hydrothermal cycle. For in situ testing, concrete samples have been placed in the large block test at Fran Ridge and the single-heater test in the Exploratory Studies Facility. Samples will also be placed in the drift-scale test. Observations will be made while the heaters are operating; after the tests, samples will be collected and follow-up studies conducted.

The drift-scale test was analyzed using a three-dimensional model. The model was used to predict (a) the maximum expected temperature rise at selected locations in the thermal test area, (b) the ventilation requirements for the neighboring drifts, and (c) the insulation requirements for the thermal bulkhead that separates the heated and unheated parts of the heated drift. The results of the modeling will be used in the design and construction of the test.

Performance Assessment Program

Current efforts are focused on preparing for the next total system performance assessment, planned for the 1998 time frame, in support of the viability assessment. Much effort was concentrated on model abstraction and testing of process-level models. In addition, experiments and modeling continued on waste package and engineered barrier materials and waste forms. A waste retrievability requirements study began; this study is a precursor to the design activity expected next period. In addition, a performance confirmation plan is being prepared.

Model Abstraction. The Project has adopted an abstraction-testing process as one measure to ensure that the total system performance assessment supporting the viability assessment is valid and defensible. The performance assessment supporting the viability assessment will be constructed of models developed to represent processes and features of both the natural and the engineered systems. The abstraction-testing process is being used to ensure the results from the highly detailed and computationally intensive site and engineered system models are properly reflected in the abstracted models used for performance assessment.

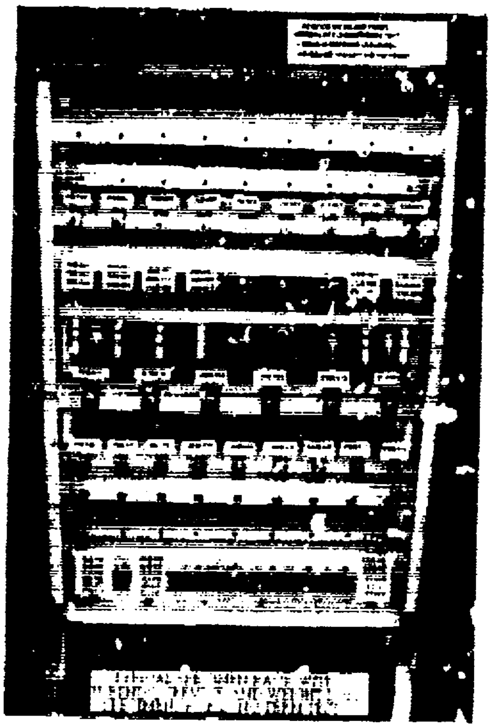
Although the processes and features are strongly interdependent, the performance assessment analysts have broken the processes into components to facilitate analysis and the abstraction-testing process. These processes and their key outputs in abstraction are shown in the box on this page.

The abstraction process includes three major elements: planning, workshops, and implementing abstraction-testing plans developed during the workshops. The workshop participants include data collectors, process modelers, subsystem performance assessment modelers, and total system performance assessment modelers.

Process	Key Output in Abstraction
Unsaturated zone flow	Percolation and seepage flux
Thermohydrologic flow	Humidity temperature
Near-field environment	Sorption, dispersion
Waste package degradation	Containment time
Waste form alteration and mobilization	Solubility, diffusive/advective flux
Unsaturated zone transport	Advective velocity distribution
Criticality	Probability, effects
Saturated zone flow and transport	Dilution
Biosphere	Dose conversion factors

This reporting period, workshops were held on the following topics: (1) unsaturated zone flow, (2) thermohydrologic flow, (3) near-field environment, (4) waste package degradation, (5) waste form alteration and mobilization, (6) unsaturated zone transport, and (7) criticality. The results of these seven workshops include a listing of the performance measures or prioritization criteria against which all issues were ranked for importance to postclosure performance. Also, a list of the highest priority issues was defined and a short synopsis of the various testing and analysis plans was developed. After the workshops, work began on the detailed testing and analysis plans. Two more workshops are scheduled for next period: (1) saturated zone flow and transport and (2) biosphere.

Abstraction-testing workshops were held on seven topics this reporting period.



Testing in the long-term corrosion facility

Materials Testing. Many different types of material studies are being conducted to investigate the potential corrosion of waste package container materials placed in the potential repository horizon. Studies in the long-term corrosion testing facility, which began shortly before the end of last period continued. Three classes of materials are being studied: corrosion resistant, corrosion allowance, and intermediate corrosion resistant. This comprehensive test is planned to last at least five years, with test specimens periodically removed and inspected. The first set of specimens was withdrawn at the end of March 1997. The information from these tests will be used to refine process models that will be provided to the performance assessment and the waste package design groups.

Investigations into conditions that increase the susceptibility of metals to aqueous film corrosion began to provide clues about the mechanics of corrosion on materials contaminated with salts typical of the repository environment. Investigators are particularly concerned about corrosion once conditions become damp and humid. This period, studies showed that salt-covered specimens of carbon steel corrode very fast at first, but with time, the salt is "consumed" by the oxidation process and the corrosion eventually stops. Later, the oxide transforms into a stable oxide and spalls off the surfaces.

In electrochemical tests, various alloys were tested to determine the potential for localized corrosion, such as pitting. The alloys were tested in brines of various salt content. Alloy C-22 (a nickel-chromium-molybdenum alloy) and Ti Grade-12 (a titanium-base alloy) were immune to localized attack under all experimental conditions tested, thus suggesting their suitability for use as inner container materials.

In other testing, galvanic corrosion tests began in January 1997 to investigate the electrochemical interaction between the dissimilar metals proposed for the multibarrier waste package designs. Tests to determine if corrosion is enhanced by microorganisms present in the potential repository were extended from room temperature to 50°C. Stress-corrosion crack growth tests are being performed in a very severe environment used to distinguish between candidate materials. Results indicate that Alloy 825 (an iron-nickel-chromium-molybdenum alloy), which is a potential candidate for the inner barrier, became susceptible to stress-corrosion cracking after being in an acidified salt solution at 90°C for 30 to 90 days.

In the abstraction-testing workshop on waste package container degradation, some of the individual models for the different corrosion modes were consolidated. This was particularly true for those affecting the inner barrier material and its interaction with the corrosion products and remaining structure of the outer barrier.

Waste Form. Waste form activities included evaluations of commercial spent nuclear fuel and defense high-level waste glass dissolution and leaching, spent nuclear fuel oxidation, and thermodynamic data development for geochemical modeling. Tests on spent nuclear fuel included flow-through tests, and testing of commercial spent nuclear fuel in three types of unsaturated zone conditions: high drip rate, low drip rate, and vapor tests. Dry-bath weight gain tests are in progress to determine the oxidation response of spent nuclear fuel. Long-term unsaturated tests (drip tests) continued on two glass compositions (Savannah River Defense Waste Processing Facility and West Valley ATM-10). A glass alteration model has been developed for use in waste package performance modeling and is reported in the Waste Form Characterization Report.

A glass alteration model has been developed for use in waste package performance modeling

In the abstraction-testing workshop related to waste form (waste form degradation and radionuclide mobilization), the highest ranked issues for spent nuclear fuel were dissolution/alteration rate, release rate, solubility limits, colloidal kinetics, and cladding degradation. High burnup spent nuclear fuel test samples were also identified as an issue. For high-level waste glass, the highest ranked issues were dissolution/alteration rate, release rate, solubility limits, and colloidal kinetics.

Waste Retrievability. A waste retrievability study is being conducted to develop the technical rationale for the mined geologic disposal system design approach to be used for complying with the 10 CFR Part 60 requirements related to retrievability. This study will also identify potential scenarios concerning the final disposition of the retrieved waste. The Retrievability Strategy Report was completed in April 1997, and the related mined geologic disposal system retrieval design activity is scheduled to be completed at the end of fiscal year 1997.

Both the retrievability system study report and the related mined geologic disposal system retrieval design activity are scheduled to be completed next reporting period

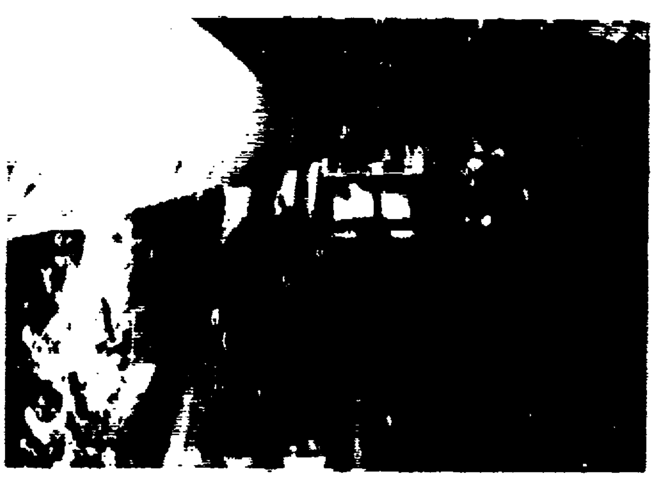
Performance Confirmation. A performance confirmation plan is being prepared. The plan will provide details of planned performance confirmation activities, including surface-based parameter evaluations, evaluations of model predictions, and corrective actions if necessary. As in the previously developed concept study report, the plan will (a) identify the processes to be simulated for postclosure performance assessment in support of a license application, (b) list the site and mined geologic disposal system design parameters needed for these analyses, (c) from this list, recommend the parameters that need to be measured, monitored, observed, tested, and analyzed following the submittal of a license application to construct a repository, and (d) describe specific performance confirmation activities and facilities for performance confirmation data acquisition and related evaluations.

Exploratory Studies Facility Design and Construction

Activities associated with the Exploratory Studies Facility concentrated on completing the south ramp excavation and testing alcoves. Only surface support facilities necessary to support subsurface construction are being built. Construction of underground support and utility facilities continued at a rate needed to support the progress of the tunnel boring machine.

Very blocky ground reduced tunnel boring machine progress in December and January to 2.5 meters per excavation day

Exploratory Studies Facility. The tunnel boring machine is now tunneling up the south ramp and at the close of the reporting period was 132 meters from reaching the south portal and daylight. During the first two months of this reporting period, the tunnel boring machine advanced at an average rate of 21.6 meters per excavation day. Very blocky ground, however, was encountered in late November that significantly reduced progress, and during December and January the average advance rate fell to 2.5 meters per excavation day. The poor ground conditions required extensive material be removed by hand from around the tunnel boring machine before invert segments were installed. Almost continuous installation of steel sets for ground support was also required.



Northern Ghost Dance Fault Alcove construction

In early February, ground conditions improved significantly so that the tunnel boring machine advance rate averaged 20 meters per excavation day for the month. Under the present schedule, the tunnel boring machine is expected to reach daylight in April.

Alcove Construction. Three test alcoves were under construction this period. Excavation of the Thermal Testing Facility was completed in early February 1997 fulfilling a Project milestone. Construction is continuing in the Thermal

Testing Facility to support the planned drift-scale test scheduled to start in the first quarter of fiscal year 1998. Excavation of the access drift of the Northern Ghost Dance Fault Alcove was extended to its planned length of 134 meters, and the exploratory borehole was reentered and extended to locate the fault and testing on the fault began. The excavation of the drill-test room for the Northern Ghost Dance Fault Alcove began. In addition, the initial phase (about 143 meters) of the drift access for the Southern Ghost Dance Fault Alcove was completed.

Design work continued to support the Thermal Testing Facility and Northern Ghost Dance Fault Alcove. Designs for the excavation for the drift-scale heater test and associated test support features and the drill-test room for the Northern Ghost Dance Fault Alcove were issued for construction.

Programmatic Activities

Project planning and licensing activities were focused on activities to support the viability assessment of Yucca Mountain as a permanent geologic repository.

__**Planning**__. The Project baselined a revision to its long-range plan in December 1996. The revision reflected the detailed fiscal year 1997 work scope and funding plan that was baselined on September 30, 1996. The long-range plan revision reaffirms the essential schedules, milestones, and key Yucca Mountain Site Characterization activities described in the revised Program Plan. The activities directly supporting the fiscal year 1998 viability assessment were planned in detail, consistent with the baselined long-range plan.

The Project baselined a revision to its long-range plan in December 1996

__**Regulatory Activities**__. Interactions with Commission staff and other organizations continued. Issue resolution activities were the [illegible] focus of several interactions with Commission staff. These interactions centered on defining the methodologies used to address the issues of seismic hazards, igneous activity, and criticality.

In response to Commission staff comments, the Department revised the first seismic topical report and resolved Commission staff comments on the second seismic topical report. Both reports are expected to be reissued next period. The Department plans to prepare the third and final seismic topical report in fiscal year 1998.

Two major interactions on igneous activity occurred this period. First, Commission staff provided three comments on the Department's process to complete the expert elicitation for the report of the Probabilistic Volcanic Hazard Analysis for Yucca Mountain. The report documented the results of the expert elicitation to assess the probability of disruption of the potential repository by an igneous eruption or intrusion, and also quantified the uncertainties associated with this assessment. The staff concluded that the elicitation process generally appeared consistent with the Commission's Branch Technical Position. The Commission staff also indicated that a path to resolve their comments on the use of expert judgment to supply licensing information has been addressed.

The Department and Commission staffs agreed on a path forward to resolving the igneous activity issue

In the second interaction, Department and Commission staffs agreed on a path forward to resolving the igneous activity issue. At the completion of the exchange, they developed a list of agreements that will help resolve the issue.

Regarding the criticality issue, the Commission staff provided comments on the Disposal Criticality Analysis Methodology Technical Report. This report provides a preliminary description of the proposed risk-based disposal criticality analysis methodology. The staffs met to discuss and begin to resolve some of the comments. Revision 1 of this technical report is expected to be completed by the end of fiscal year 1997 and will provide the basis for a topical report that will seek Commission acceptance of the proposed methodology that will include the technical basis for burnup credit.

The Department completed the first part of the second phase of the transition of quality assurance functions from the affected organizations to the Department

Quality Assurance. In quality assurance activities, the Department completed the first part of the second phase of the ongoing reengineering of the quality assurance function within the Program. This effort began in response to initiatives to improve the effectiveness of work in Departmental offices and also to cut costs. The implementation of this part of the reengineering plan was controlled by a transition plan that transferred the surveillance function performed by the individual affected organizations to the Department, effective in February 1997. The Department intends to implement the final phase to transition the remaining quality assurance functions to the Department during the next reporting period. In addition, in March the Department issued Revision 6 of the Quality Assurance Requirements and Description document, the principal Program quality assurance requirements document. This revision addressed and incorporated Commission comments concerning scientific investigation and design control.

Program Outreach. In Program outreach activities, Project staff completed a video that highlights Project activities completed during the last fiscal year. Also updated was the Project stakeholder plan that describes plans for fostering public involvement in Project activities.

Numerous educational activities continued, including operation of the Yucca Mountain Science Centers, scouting workshops, Science Bowl competition, Scout Expo '96, and a Futures Expo on career opportunities. Project personnel participated in ceremonies opening the Net Day '96 initiative in Nevada. This initiative was to connect schools to the Internet and had been supported by Project personnel through contributions and volunteer effort. For the Beatty Science Center, the Project compiled an album of historic photos of early native Americans for use in new cultural resource activities.

Activities at a science center

EPILOGUE

On April 25, 1997, the Project completed the excavation of the approximately five-mile tunnel of the Exploratory Studies Facility, which had begun in September 1994. This is a major Project milestone, and the completed tunnel serves as an underground laboratory providing scientists and engineers the opportunity to gather information not readily available from any other source. Scientific experiments in the tunnel and alcoves continue to provide valuable information to support the viability assessment and future activities such as the site recommendation and license application. Testing in the Thermal Testing Facility alcove of the tunnel also continued as the single-heater test moved into the cooling phase of the experiment. The nine-month heating phase was completed on May 28, 1997. The cooling phase will continue for approximately nine months. Data collected during the heating and cooling phases will be analyzed and compared with model results.

The Department extended for a third time the public comment period for its rulemaking activity for the proposed revision of its siting guidelines (10 CFR Part 960). The public comment period closed May 16, 1997. The Department expects to issue a final rule in mid-late 1998.

In June 1997, the Department approved a modification to the current fiscal year and long range plan to include the construction of an east-west cross drift. The effort will include design and construction of the drift, construction of three test alcoves, and drilling and testing supporting these excavations. Data from this construction will enhance scientific understanding of the behavior of the site, including engineering, construction, health, safety, regulatory, and performance aspects of the potential repository.

The Department is considering a revised format for the Progress Report that would provide a more succinct, high-level summary of the most important activities, or focal points of the Program. The revised format would include brief discussions of major activities, focus on items of greatest interest during the reporting period, and result in a significantly streamlined document. In generating this document, increased reliance would be placed on references for technical details and Appendix A would be the first reference to the Progress Report. Appendix A would be maintained as a separate referenced document and be revised once every six months. Provision of a streamlined Progress Report would provide readers with a more user-friendly document and meet the reporting requirements of the Nuclear Waste Policy Act and the Nuclear Regulatory Commission's 10 CFR Part 60.

PROGRESS REPORT #16

CHAPTER 1 - INTRODUCTION

This Site Characterization Progress Report summarizes progress on site characterization activities at Yucca Mountain, Nevada, for the period October 1, 1996, through March 31, 1997. The report is the sixteenth in a series issued approximately every six months to report progress and results of site characterization activities being conducted to evaluate Yucca Mountain as a possible geologic repository for the disposal of spent nuclear fuel and high-level radioactive waste. This progress report is prepared in accordance with Section 113(b)(3) of the Nuclear Waste Policy Act of 1982, as amended (NWPAA, 1987), and 10 CFR 60.18(g).

This report highlights work started, in progress, and completed during the reporting period. In addition, this report documents and discusses changes to the Office of Civilian Radioactive Waste Management (OCRWM) Site Characterization Program (Program) resulting from the ongoing collection and evaluation of site information, systems analyses, development of repository and waste package designs, and results of performance assessment activities. Details on the activities summarized can be found in the numerous technical reports cited throughout the progress report.

Yucca Mountain Site Characterization Project (Project) activities this period focused on implementing the near-term objectives of the revised Program Plan issued last period. Near-term objectives of the revised Program Plan include updating the U.S. Department of Energy's (DOE) repository siting guidelines to be consistent with a more focused performance-driven program; supporting an assessment in 1998 of the viability of continuing with actions leading to the licensing of a repository; and if the site is suitable submittal of a Secretarial site recommendation to the President in 2001 and license application the U.S. Nuclear Regulatory Commission (NRC) in 2002. During this reporting period, the Project developed and baselined its long-range plan in December 1996. That revision reflected the detailed fiscal year (FY) 1997 work scope and funding plan previously baselined at the end of FY 1996. Site characterization activities have been focused to answer the major open technical issues and to support the viability assessment.

The following sections of Chapter 1 present Project history, outline the bases for the near-term objectives, discuss the DOE's strategies for achieving these objectives, and note progress in achieving these objectives. The chapters after this introduction describe Project activities that occurred during this reporting period that support the Program.

1.1 HISTORICAL PERSPECTIVE

The Project started in 1977 when the DOE began evaluating the possibility of disposing of high-level radioactive waste in a geologic repository at the Nevada Test Site. Over the next two years, the DOE investigated a number of sites near the Nevada Test Site and decided to concentrate exploration efforts on the tuffs of Yucca Mountain. In 1980, the DOE conducted a formal screening of Nevada Test Site Area 25 (within which a part of Yucca Mountain lies). This analysis was conducted to be compatible with the area-to-location phase of site screening described in the national siting plan used before the passage of the Nuclear Waste Policy Act of

1982 (NWPA, 1993). As a result of this formal evaluation, the DOE identified Yucca Mountain as a potentially acceptable site in February 1983.

The Nuclear Waste Policy Act of 1982 established a nationa[illegible] [illegible]licy for the disposal of high-level radioactive waste and spent nuclear fuel. The Act also c[illegible]d the OCRWM within the DOE and assigned that office the responsibility for developing a waste management system.

In response to requirements in Section 112(a) of the Act, the DOE issued the 10 CFR Part 960 guidelines in 1984 for evaluating the suitability of sites for repositories and started the site screening process. The Environmental Assessment for Yucca Mountain (DOE, 1986), also required by the Act, was issued in 1986. In response to the issuance of the Environmental Assessment, surface-based studies at Yucca Mountain accelerated. Initially, these studies consisted of nonsurface-disturbing testing in existing exploratory boreholes and wells controlled by the government; analyses of, and experiments with, rock and water samples, geophysical surveys, meteorological, hydrologic, and seismic monitoring, geologic mapping, and sampling of surficial materials. From the mid-1980s to 1994, the DOE planned for and then conducted a comprehensive site characterization program based on the Site Characterization Plan (SCP) (DOE, 1988).

With the 1987 Amendments to the Nuclear Waste Policy Act, Congress designated Yucca Mountain as the only site to be characterized to determine its suitability as a geologic repository. During the following year the OCRWM issued an SCP in accordance with the Act and continued conducting a program of detailed site-specific investigations and evaluations to assess the suitability of Yucca Mountain. The NRC issued its review of the SCP, the Site Characterization Analysis (NRC, 1989), in July 1989. This document identified points requiring clarification and NRC concerns in the form of comments, questions, and objections.

During the first half of FY 1994, the DOE conducted preliminary evaluations of various options for restructuring the repository program to meet changing needs and expectations. The 1988 SCP had presented a comprehensive testing, design, and performance assessment program. The scientific information obtained as the Project activities progressed was expected to be used to focus the Program on activities needed for site characterization and safety analysis. However, external and internal pressures since 1988 tended to broaden, rather than to focus the Program, and resulted in rising expectations about the level of certainty in understanding the natural geologic systems and the performance of engineered barriers in the geologic setting. By the late 1980s, Congress had begun to express concern about continuing growth in the estimated cost of site characterization. In addition, because the site characterization schedule did not require definitive results until the license application was completed in 2001, progress was difficult to demonstrate and to measure. Thus, over the last half of FY 1994, the DOE developed and refined a new approach designed to show early observable progress within the financial resources likely to be available. The 1994 Program Plan, issued December 19, 1994 (DOE, 1994a), summarized the new approach, and the DOE began implementing this new approach during the first half of FY 1995.

By the end of FY 1995, however, Program redirection requested by Congress and consequent reductions in funding for FY 1996 again significantly altered the direction of the repository program described in the 1994 Program Plan. The DOE recognized that, because of the significant reduction in funding, a much reduced repository program would be required that, consistent with Congressional guidance, focused on the core scientific activities at Yucca Mountain and that deferred the preparation and submittal of a license application to the NRC. As a result, the target dates in the 1994 Program Plan for constructing a geologic repository and for emplacing waste underground also were deferred. The funding reductions resulted in curtailed site investigations and postponement of both the environmental impact statement and revisions to the annotated outline for the license application.

During the second half of FY 1996, the DOE released Revision 1 of the Civilian Radioactive Waste Management Program Plan in draft form (DOE, 1996a). This draft plan reflected a revised Program approach to be compatible with congressional guidance and expected funding levels and to answer the most critical technical issues remaining about the design of the repository and its expected performance in the geologic setting. This modified approach reflected the increased technical understanding of the Yucca Mountain site. This draft plan, as implemented by the Project's long-range plan (CRWMS M&O, 1996a) and annual fiscal year plans, incorporates a focused, integrated program of site characterization, design, and performance assessment that will result in a license application for a repository at Yucca Mountain, if the site is suitable. The Energy and Water Development Appropriations Act for FY 1997 in effect approved the DOE's new strategy and directed that a viability assessment of the Yucca Mountain site be submitted to the President and Conrress by September 30, 1998.

1.2 OBJECTIVES OF THE REVISED PROGRAM PLAN

In its revised Program Plan, the DOE identified three near-term objectives: (1) updating in 1997 the DOE repository siting guidelines to be consistent with a more focused program driven by the results of performance analysis, (2) supporting an assessment by 1998 of the viability of continuing with actions leading to the licensing of a repository, and (3) if the site is suitable, submitting a Secretarial site recommendation to the President in 2001, and a license application to the NRC in 2002.

The following sections discuss the near-term objectives of the revised Program Plan.

1.2.1 1997 Update of Repository Siting Guidelines

A key element in the revised Program strategy is reviewing and proposing revisions to the DOE repository siting guidelines. Over the past decade, legislation has moved to characterization a single site instead of multiple sites and toward developing site-specific standards to protect the public. First, in 1987 Congress directed that the DOE characterize the Yucca Mountain site and terminate activities at the other sites. In 1992 Congress directed the U.S. Environmental Protection Agency to promulgate a site specific standard for Yucca

Mountain for protection of t public. The NRC must then modify the technical requirements and criteria for licensing a repository (10 CFR Part 60) to be consistent with that site specific standard. Thus, the DOE proposes to amend its siting guidelines (10 CFR Part 960), which were promulgated in 1984, in response to these national policy changes and to increasing technical understanding of the Yucca Mountain site.

The proposed revisions to the siting guidelines are intended to eliminate the comparative siting criteria for evaluating the suitability of the Yucca Mountain site for development as a repository. Instead, the DOE is proposing to add a site-specific subpart focused on the overall performance of the repository system during operations and after closure, rather than separately evaluating individual aspects of the site. An overall system performance ap roach is the appropriate method to consider all relevant site features because it identifies in an integrated manner those attributes of the site and engineered components that are most important to the protection of public health and safety.

The DOE is using a public rulemaking pr cess to modify 10 CFR Part 960, which began with a Notice of Proposed Rulemaking (61 FR 66157) issued for public comment. The Notice of Proposed Rulemaking published on December 16, 1996, began a public comment period that is scheduled to close next reporting period. The DOE expects to issue a final rule in mid-late 1998. When implemented, the regulatory changes should s amline the process for site evaluation and repository development, while protecting public health and safety and the environment.

1.2.2 1998 Viability Assessment

The second principal objective of the revised plan is to address by 1998 the major open technical issues, including those related to the waste containment and isolation strategy, so that an informed assessment can be made of the viability of licensing and constructing a geologic repository at the Yucca Mountain site.

The viability assessment is not the same as the site suitability determination and site recommendation. Rather, it is a step along the way that is essential for the rational completion of the site recommendation. The viability assessment has two purposes: first, to guide the completion of the work required for an evaluation of site suitability and preparation of a license application, and second, to provide the legislators with a better estimate of the viability of a geologic repository at the Yucca Mountain site.

The viability assessment is a logical convergence point at which the DOE can make an improved appraisal of the prospects for geologic disposal at Yucca Mountain using the results of the program first described in the 1988 SCP and the results from the excavation of the Exploratory Studies Facility. These results allow the DOE to develop an integrated picture of the repository system at the site that was previously impossible.

The evaluation of the site will not be finished in 1998, but the assessment will bring many individual data elements together into a comprehensive perspective for the first time. The information produced for this viability assessment will be an important validation of the practicality of the existing national policy of geologic disposal and will allow policy makers to make a measurably improved judgment of the prospects for recommending Yucca Mountain for repository development and licensing to authorize construction. When the data and information are delivered, policy makers may determine that the current program should be changed or amended or that DOE should continue to implement the revised Program Plan. Currently, the Project is concentrating most of its effort on supporting the viability assessment.

1.2.3 Site Recommendation and License Application

If the site is determined to be suitable, the DOE would proceed with issuing the statutorily prescribed environmental impact statement and Secretarial recommendation to the President. If the site were approved by the President, the DOE would submit a license application to the NRC in 2002.

Environmental Impact Statement

The Nuclear Waste Policy Act of 1982, as amended, requires that a final environmental impact statement accompany any Secretarial site recommendation to the President. Any such environmental impact statement would be adopted to the extent practicable by the NRC in connection with the issuance of a construction authorization and license. The environmental impact statement process began with the Notice of Intent published in the *Federal Register* on August 7, 1995. The public comment period closed December 5, 1995, following 15 public meetings across the nation. Because of decreased Project funding for FY 1996 and direction provided by Congress in that fiscal year appropriation, preparation of the environmental impact statement was deferred until this reporting period. Work has now begun on reviewing and summarizing public scoping comments, gathering data, and identifying data needs. The draft environmental impact statement is scheduled to be issued in FY 1999 and the final environmental impact statement issued in FY 2000.

Site Recommendation

If the site is determined to be suitable with respect to the then-existing siting guidelines, a decision to recommend to the President that the site be approved for development as the nation's first high-level waste repository is expected in FY 2001. In FY 2000, the DOE would inform the public that it is considering a site recommendation and would announce a schedule for public hearings on the possible site recommendation. After the hearings and before recommending the site to the President, the DOE would notify the State of Nevada about the decision to recommend the site.

As required by Section 114 of the Nuclear Waste Policy Act of 1982, the basis for the decision on the site recommendation would include (a) a description of the proposed repository, including preliminary engineering specifications; (b) a description of the waste form or packaging and an explanation of the relationship between the geologic environment and the waste packages and waste forms; (c) a discussion of data obtained during site characterization relating to the safety of the site; (d) the final environmental impact statement; and (e) preliminary comments by the NRC concerning the sufficiency of site characterization. The site recommendation also would include the views and comments of the Governor of the State of Nevada, the State legislature, and any affected Indian tribe, as well as other information the Secretary of Energy considers appropriate and any impact report submitted by the State.

A Project Integrated Safety Assessment, scheduled to be completed in 1998, will present integrated information about the technical elements of the Program. This document will describe and integrate information on site conditions, repository and waste package design, and performance assessment. After completion, the document will be provided to the NRC for review as one basis for its preliminary comments on the sufficiency of information on the site and design for inclusion in a license application. The Project Integrated Safety Assessment will be used as the starting point for developing the license application.

License Application and Revised Approach to Licensing

The goal of submitting a successful license application remains central to the Program's mission. The technical reports and supporting data associated with the components of the viability assessment will help improve the understanding of the repository concept, and provide a comprehensive appraisal of the prospects for licensing and constructing a geologic repository at Yucca Mountain. Although the information will not be sufficient for licensing, this work is a logical step toward developing a first-of-a-kind repository. In completing the license application, the Project will define a repository concept that includes a facility and waste package design consistent with the characteristics of the Yucca Mountain site and will assess the performance of this repository. Appropriately, the DOE will develop a repository concept and ensure that it adequately protects public health and safety before seeking approval from outside parties. In so doing, the Project will examine alternatives and will propose a repository system that can be achieved within rational cost and schedule restraints.

During this reporting period, the DOE began developing three products that will support development of the license application: the License Application Plan, the License Application Management Plan, and the Technical Guidance Document for License Application Preparation. As described in Section 1.2.2, the License Application Plan is one of the products of the viability assessment and will describe the work to be performed between the viability assessment and the license application.

The License Application Management Plan will provide the management framework within which the license application will be developed. Information included will be the layout of the license application, the document control process for the license application, and a step-by-step description of the process by which the license application will be developed. The management plan is expected to be completed by the end of FY 1997.

The Technical Guidance Document for License Application Preparation will provide guidance to authors on the technical and licensing content of their license application chapters. The technical guidance document will list the regulatory requirements applicable to each chapter and will provide acceptance criteria for the authors to use to verify they have provided an adequate licensing case in their chapters. The document will also describe how industry standards and other available guidance should be used in the licensing case to be presented in the license application. The initial guidance document is scheduled to be completed by the end of FY 1997. At that time, some of the regulatory requirements and acceptance criteria will still be under development, so this initial document will still have gaps and will provide interim guidance.

The DOE believes that interactions with the NRC staff before the license application should focus on two objectives: reaching a common understanding regarding the issues that are significant to overall repository performance, and reaching agreement on the adequacy of proposed methodologies and approaches to address important technical issues such as criticality control and seismic design. The goal is to reach a mutual understanding of the repository concept as it develops. This understanding will provide a basis for NRC preliminary comments (to be included in the DOE site suitability package) on the sufficiency of site characterization and design for inclusion in a license application.

This approach is a departure from previous efforts that focused on resolving individual issues related to specific site characteristics in isolation from one another or from a specific design concept. The DOE believes the sufficiency of site characterization data and analyses can only be determined within the context of a coherent repository concept that includes both design and system performance. Thus, the DOE will first develop the overall repository concept before addressing specific issues related to licensing. The DOE will seek insights from the NRC staff throughout this process regarding issues affecting licensing and approaches and methodologies for addressing specific technical issues. This reporting period, interactions continued with the NRC staff on the issues of seismic hazards, igneous activity, and repository criticality.

Since the NRC's 1989 revision of 10 CFR Part 2, including a new Subpart J that required the DOE to design and develop an electronic information management and distribution system (designated the Licensing Support System) was issued, discussions with the NRC have led to an understanding that most of the requirements for a Licensing Support System may be met by using available Web technology. The NRC and DOE are actively pursuing alternative solutions to a Licensing Support System. While the NRC is proposing reasonable alternatives to a Licensing Support System in a rewrite of Subpart J, Project staff is reprocessing records into a format that will allow access via the Internet or transfer to a Licensing Support System if that requirement remains and prototyping an electronic format for licensing documents. The prototype effort is expected to conclude in June 1997 and will allow the DOE to determine whether electronic licensing documents are feasible and the manner in which such documents should be generated.

Linkages from a licensing document to its supporting documentation would provide easy traceability for reviewers.

The current strategy recognizes that additional information would be obtained through continued surveillance, measurement, testing, analysis, and performance assessment during the construction and operation of the repository. This information would be reflected in subsequent license application updates and amendments.

Beyond license application submittal, the Project would place high priority on those tests designed to enhance confidence about long-term performance as part of the performance confirmation program required under 10 CFR Part 60, Subpart F. During this time, the long-duration in situ tests that measure moisture redistribution and changes in rock properties in response to thermal loading would provide key information. The Project also would give high priority to confirming the behavior of engineered barriers within the range of expected repository conditions.

If the NRC issues a license to receive and possess spent nuclear fuel and high-level radioactive waste, the scientific work will focus on verifying the terms and conditions of the license with regard to site and repository characteristics and performance and on obtaining data on changes to the site caused by repository construction and waste emplacement. The object would be to confirm, in accordance with 10 CFR 60.140(a), the basis for earlier predictions about containment and isolation, to confirm that natural and engineered systems are functioning as expected, and to test the models that will be relied upon for confirming the long-term predictions required to support a decision by NRC to permit closure of the repository. The application for a license amendment to close the repository will not be submitted until sufficient confirmatory test information is available to provide adequate confidence to support a decision to close the repository. The repository will be designed and operated to preserve the option to retrieve the emplaced waste for up to 100 years after the beginning of waste emplacement, or until the NRC decides to permit permanent closure. A decision to exercise the retrieval option may be made on the basis of the results from performance confirmation, or it may be prompted by a policy decision related to geologic disposal of spent nuclear fuel, or recycling of fissile material in nuclear reactors.

1.3 WASTE CONTAINMENT AND ISOLATION STRATEGY

The waste containment and isolation strategy is being used to focus the remaining effort to the viability assessment and to guide the work beyond the viability assessment. The strategy focuses on two technical objectives: first, to limit the annual dose to members of the general public following permanent closure of the repository; and second, to provide total containment of the waste within the emplaced waste packages for thousands of years during the period of highest radionuclide inventory and temperature. The strategy outlines the approach to addressing and resolving postclosure performance issues for licensing and also focuses the science and design work needed to determine postclosure performance in the period leading to the viability assessment and beyond. The strategy also incorporates new site information and designs, realistic performance predictions, and potential regulatory changes.

A summary version of the strategy was completed and distributed to the Nuclear Waste Technical Review Board and to the NRC in July 1996. After the summary was completed, new information derived from interpretations of several site investigations indicated the potential for average flux values at the repository horizon in the range of 1 to 10 millimeters per year or even higher, with a part of this flux associated with fast pathways. Percolation flux affects all the system attributes identified in the strategy as most important for predicting the performance of the engineered and natural barriers: rate of water seeping into the repository, waste package lifetime (containment), rate of release of radionuclides from breached waste packages, radionuclide transport through engineered and natural barriers, including dilution in the saturated zone below the repository. Higher percolation flux could mean shorter travel times to the accessible environment, an increase in the number of waste packages being dripped on, higher relative humidity, and a reduction in the ability of the heat from the waste to drive water in the host rock away and to produce and maintain a dry repository. Determining percolation flux is particularly challenging because no method is available to measure these low flux values. Also, empirical data are scarce and there is a need for further study. Thus, the DOE began re-evaluating the strategy. A revised summary is scheduled to be issued next reporting period.

The percolation flux information has far-reaching effects, impacting design, performance assessment, and cost—the components of the viability assessment. Thus, the evolving flux information and unsaturated zone models will be factored into the ongoing work toward viability assessment, site recommendation, and license application.

The viability assessment will include:

1. The preliminary design concept for the critical elements for the repository and waste package;

2. 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;

3. A plan and cost estimate for the remaining work required to complete a license application; and

4. An estimate of the costs to construct and operate the repository in accordance with the design concept.

The following sections describe these deliverables and report current status.

Repository and Waste Package Design

The first component of the viability assessment— repository and waste package design—addresses those design elements critical to determining the feasibility and performance of the repository and engineered barrier system. The effort will evaluate the technological feasibility of the designs but will not develop all the detail needed for licensing.

The repository and waste package designs build on previous work, including the Mined Geologic Disposal System Advanced Conceptual Design Report (CRWMS M&O, 1996b) that was published in March 1996. The Project continues its philosophy of a phased and evolving design that will support the viability assessment, environmental impact statement, site recommendation, and license application. Current design is focusing on designing systems, structures, and components that have little or no regulatory precedent and have a major impact on performance assessment, schedule, constructability, and cost. These issues, based on the waste containment and isolation strategy, arise from thermal management of the waste-generated heat, the role of supplemental engineered barriers, corrosion of waste packages, and dissolution of radioactive wastes.

Scientific and engineering information obtained during testing and analysis of data from the Exploratory Studies Facility will be incorporated into the designs. Site programs and performance assessment activities will provide data and criteria for designs. The designs will serve as a basis for estimating repository costs and schedules and for identifying additional design work needed for licensing. The design information will also be used as an input to the total system performance assessment that will support the viability assessment.

This period in keeping with its design philosophy, the Project began developing guidance to support its phased design approach toward licensing. Design activities this reporting period include modifying waste handling operations to reflect the de-emphasis of the multi-purpose canister and performing design analyses (including structural, thermal, shielding, and criticality) to accommodate uncanistered spent nuclear fuel. In addition, laboratory tests are being performed on waste package materials and waste forms to provide input for waste form degradation process models.

Total System Performance Assessment

The second component of the viability assessment—total system performance assessment—will describe the probable behavior of the repository system consisting of the natural and engineering systems and calculate the variation and uncertainty in this performance.

The performance assessment, scheduled for the 1997-1998 time frame, will reflect an integrated site and engineered system using design concepts and data available at that time. Uncertainties will continue to exist in the characterization of both the engineered component and natural system processes. These uncertainties will be reflected in the alternative conceptual models and parameter distributions considered in the analyses. Thus, the performance assessment also will evaluate the possible range of performance caused by uncertainty in key factors, such as ground-water flow, thermal effects, and corrosion.

To ensure that the bases for the assessment are valid and defensible, the Project has developed a process to perform an integrated total system performance assessment to support the viability assessment. Model development must be focused on issues that are most important to performance. To ensure traceability (a complete and unambiguous record) and transparency (ease of understanding by the reader and reviewer), the bases for assumptions and results must be clear, readable, complete, and documented. Traceability and transparency are continuing concerns of both the Project and the NRC, and these concerns were also raised by the Performance Assessment Advisory Group of the Nuclear Energy Agency. This group compared and evaluated the specific technical approaches of ten recent system assessments, including two that address Yucca Mountain (one by DOE and one by NRC). In their review, the group noted problems in traceability and transparency.

As a critical step to ensure traceability, a process was established to ensure that performance assessments properly reflect results from the highly detailed and computationally intensive site and engineered system models and from the scientific data that constitute the basis of these models. The activities are designed to integrate the work of site characterization, design, environmental programs, and performance assessment. It is not feasible nor efficient to incorporate all the complexity inherent in all the specific process models into a probabilistic total system performance assessment calculation. Instead, abstracted models are used as surrogates for the comprehensive process models. The abstracted models must, however, maintain the essential elements of the process model, including key interdependencies. Several processes have been identified as key: unsaturated zone flow, thermohydrologic flow, waste package degradation, near-field environment, waste form alteration and mobilization, unsaturated zone transport, saturated zone flow and transport criticality, and biosphere. This abstraction process is critical to the success of the Project.

For the total system performance assessment supporting the viability assessment, Project staff have undertaken an extensive program of defining, developing, and testing abstracted models in the nine technical disciplines most important to repository performance. The abstraction and testing activities occur in three steps. The first step is the planning needed to identify a preliminary list of issues and the activities to be accomplished. The next step is to hold a workshop to develop a consensus on the relative importance of issues related to the process model and to develop plans to analyze the highest ranked issues. The workshops include data collectors, process modelers, subsystem performance assessment modelers, and total system performance assessment modelers. Following the workshops, the third step occurs when the details of the abstraction-testing plans are completed and the analyses and testing performed.

This reporting period, seven workshops were held, and two more workshops are scheduled for next period. The abstraction workshops are only one of several measures the Project is taking to ensure the traceability and transparency of the total system performance assessment supporting the viability assessment. Two other measures, plus their accompanying records, are being used to ensure traceability

First, a formal expert elicitation process has been instituted. In support of the assessment for the viability assessment, this process will follow the nine-step process outlined in the NRC's Branch Technical Position on the use of expert elicitation (NRC, 1996a). The purpose of this process is (a) to quantify and document the uncertainties in the process models to strengthen the assessment and (b) to focus on process models that are very significant to total system performance. This process is currently under way on the site-scale unsaturated zone flow model—one of the most important of the foundational process-level models. A series of three workshops involving data collectors and analysts, modelers, and nationally recognized experts in the field of unsaturated zone flow characterization was conducted.

Second, a peer review panel of external experts has been established to monitor and review the preparations for the assessment, as well as the final product itself. The peer review panel will first review the previous total system performance assessments from 1991, 1993, and 1995 and make observations on the plans, approach, and assumptions for the assessment to support the viability assessment. The reviewers will also review the process modeling and the abstraction process. Finally, they will provide a formal peer review of the assessment supporting the viability assessment and the comments and recommendations will be incorporated into the assessment that will support the license application. During this reporting period, the peer review panel was convened and its orientation began.

Besides the steps to ensure traceability, the Project is also using specific measures to ensure transparency. An initiative is starting to examine ways of presenting the total system performance assessment results graphically so that they would be more easily understood by those who are not performance assessment specialists. The Project is also investigating using hypertext to increase the reviewer's electronic access to data sources and cited materials and developing a computerized data retrieval and selection system to help trace and document decisions made in data selection.

License Application Plan

The third component of the viability assessment—license application plan—will define the work required to complete a license application. Submittal and docketing of a license application to the NRC, should the Yucca Mountain site be found suitable, is the DOE's central goal.

During this reporting period, Project staff began developing the License Application Plan, which will guide the development of the license application. The plan will describe Project work to be performed between the viability assessment and the license application, and give a schedule and a cost estimate for this work. The plan will also outline the Project's licensing strategy, describe the license application content requirements specified by 10 CFR 60.21, and describe the performance confirmation plan. The information developed for the License Application Plan will also be used to support the development of the license application, if the site is determined to be suitable. The draft plan is scheduled to be completed late in FY 1997.

Cost and Schedule Estimates

A set of estimated costs and schedules for the repository system is the fourth component of the viability assessment. The estimates will encompass completion of site characterization, 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 licensing and construction of a geologic repository.

The estimates will be based on the repository and waste package designs developed for the viability assessment and on scientific testing and analyses completed by 1998. Because not all the details of the design will be developed fully by the 1998 viability assessment, some design assumptions will be used to complete the estimates.

Work began this period on developing the cost to construct and operate a repository. The cost estimating process includes the development of a cost analysis document that will summarize the assumptions to support the cost estimate, the development of a life cycle cost schedule, and the identification of models to be used in the estimate. The cost analysis document will also define data bases and data sources used and will contain a draft cost estimate for review. This reporting period, the Project generated an annotated outline for the cost analysis document, updated the cost account structure, updated cost models, and developed a life cycle cost schedule. Work is continuing on assembling model descriptions for each cost module, assembling assumptions, collecting and integrating data, and generating a draft of the cost analysis document summarizing the FY 1997 work.

Progress Report

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1.4 PROGRESS REPORT CONTENT

As shown in the box on Page 1-13, the main chapters of the progress report deal with programmatic, site, design and construction, and performance assessment activities. Various appendixes support the main text. This report reflects the information available and the status of the Project as of March 31, 1997. An Epilogue has been added at the end of the Executive Summary to identify important events occurring after the close of the reporting period and before the report is printed.

In addition, for more general readers, the report begins with a Note to Readers that summarizes major achievements during the reporting period. The Executive Summary is a summary of major decisions, accomplishments, and issues of interest during the reporting period.

CHAPTER 2 - PROGRAMMATIC ACTIVITIES

This chapter reports the results of the Yucca Mountain Site Characterization Project (Project) planning and regulatory activities, including baseline control, environmental compliance, and licensing. Other programmatic activities reported include those in quality assurance (QA) and public outreach. Programmatic activities focus, evaluate, plan, control, and ensure the quality of site characterization, design, and performance assessment activities.

Programmatic activities this period focused on supporting two of the three near-term objectives of the U.S. Department of Energy's (DOE) revised Program Plan: updating in 1997 the regulatory framework for determining the suitability of the site for the proposed repository concept and providing information for a 1998 viability assessment of continuing toward the licensing of a repository (DOE, 1996a). The key features of the revised Program Plan, as well as DOE's philosophies for achieving its near-term objectives, are discussed in Chapter 1. The following summarizes the notable programmatic accomplishments during this reporting period.

In line with the goals of the Plan, the DOE published a Notice of Proposed Rulemaking on December 16, 1996 (61 FR 66157), to amend its siting guidelines in 10 CFR Part 960 by adding a new site-specific subpart for Yucca Mountain. Public comments are currently being taken, and the public comment period is scheduled to close next period. In other planning and control activities, the Project baselined a revision to its long-range plan. The revision reflected the work scope and funding plan for fiscal year (FY) 1997.

In regulatory activities, the U.S. Nuclear Regulatory Commission (NRC) staff and DOE identified several points of agreement that define a path for resolving the issue of igneous activity. Work began on three documents that will support the development of the license application: the License Application Plan, the License Application Management Plan, and the Technical Guidance Document for Preparation of the License Application.

Additional progress was made in reengineering the QA function within the Office of Civilian Radioactive Waste Management (OCRWM) Site Characterization Program (Program). This is a phased transition of QA functions from the individual affected organizations to the OCRWM Office of Quality Assurance; the transition is expected to be complete in the next reporting period. Once complete, the Office of Quality Assurance would have complete responsibility for all QA functions within the OCRWM Program.

Details of these and other programmatic activities are presented in the rest of the chapter.

2.1 PROJECT PLANNING AND BASELINE CONTROL

The Project baselined a revision to its long-range plan and also began implementing the recommendations for streamlining the document hierarchy. The Mined Geologic Disposal System (MGDS) Requirements Document is currently being revised, and when approved, that revision will affect several of the Project regulatory and management control documents.

2.1.1 Project Planning and Scheduling

The Project baselined a revision to its long-range plan in December 1996. That revision reflected the detailed FY 1997 work scope and funding plan that was baselined on September 30, 1996. The long-range plan revision reaffirms the essential schedules, milestones, and key Yucca Mountain site characterization activities described in the revised Program Plan.

Detailed annual planning for FY 1998 work scope is being carried out in stages. The activities directly supporting the FY 1998 viability assessment were planned in detail during this reporting period using an integrated schedule and detailed basis-of-estimates (a correlation of the scope of work and the resources required to complete the work). The detailed planning is consistent with the baselined long-range plan, although some schedule adjustments have been made. No major Project milestones have been negatively impacted.

Forecast: The remainder of the FY 1998 activities will be planned, but not to the level of detail as those activities directly supporting the viability assessment. Instead, a compilation of all activities will be costed, analyzed, and prioritized. Some adjustments may be made to the detailed scope or schedule to accommodate required work interfaces and to remain within the administration's budget submittal. This work will support a further update to the Project long-range plan that will also address results of Project work accomplished to date. During the last quarter of FY 1997, the detailed Project annual plan for FY 1998 will be prepared and baselined.

2.1.2 Document Hierarchy and Program Baseline

The Civilian Radioactive Waste Management System (CRWMS) Document Hierarchy for OCRWM identifies both Program- and Project-level documents, including both regulatory and management documents. During this reporting period, streamlining of the document hierarchy, as recommended by the Technical Baseline Hierarchy Task Group, was approved, and implementation began. The recommendations of the task group were incorporated into baseline change proposals. The changes incorporated included (a) moving the MGDS Requirements Document from Program-level control to Project-level control; (b) replacing the existing repository and engineered barrier requirements documents with the MGDS Requirements Document; and (c) moving the repository and engineered barrier requirements documents to the CRWMS Management and Operating Contractor (M&O) control until the system design documents are approved and controlled by the Change Control Board. Implementing this streamlining of the document hierarchy will directly affect the requirements documents listed in Appendix B.

2.1.2.1 Regulatory and Management Controls

Changes in Project-level documents that control both regulatory and management activities were made during this reporting period. The MGDS Requirements Document (DOE, 1996b) was revised to coincide with Revision 3 of CRWMS Requirements Document and the

Technical Baseline Streamlining Initiative. Revision 3 of the MGDS Requirements Document is in review. The Controlled Design Assumptions Document (CRWMS M&O, 1996c) is being revised to incorporate key design assumptions supporting the viability assessment. Changes to several documents, however, are on hold pending the completion of the current revision of the MGDS Requirements Document.

Forecast: When Revision 3 of the MGDS Requirements Document is approved, the repository and engineered barrier requirements documents will be removed from Level 2 Change Control Board control. The changes in Project-level documents that control both regulatory and management activities as identified in the current CRWMS Document Hierarchy are summarized in Appendix B.

2.1.2.2 Change Control Board Actions

The mission of the Project Change Control Board is to ensure that changes to the Project baseline or documents controlled by the Change Control Board are made with adequate consideration of the technical, regulatory, QA, programmatic, and cost and schedule impacts that such a change would have on each element of the Project. Change control, as exercised by the Change Control Board, prevents unnecessary, untimely, or marginal changes and expedites the approval and implementation of changes that are needed to significantly benefit the Project.

Appendix C provides a table summarizing significant changes presented to the Project Change Control Board since October 1, 1996. The changes exhibit cost, schedule, and work scope activities that impacted the cost and schedule baseline.

Forecast: Changes to regulatory and management control documents will proceed following the completion of the current revision of the MGDS Requirements Document.

2.2 REGULATORY ACTIVITIES

Regulatory activities associated with suitability, environmental compliance, and licensing (including interactions with the NRC) continued this reporting period to support the near-term objectives of the revised Program Plan.

2.2.1 Suitability Activities

The DOE published a Notice of Proposed Rulemaking in the *Federal Register* on December 16, 1996, (61 FR 66157) to amend its siting guidelines in 10 CFR Part 960 by adding a new site-specific subpart for Yucca Mountain. A public hearing was held in Las Vegas, Nevada, on January 23, 1997, as part of the public comment process. In response to comments received from the public, the original 60-day public comment period that began on December 16, 1996, was extended to 91 days. This extension was announced in a second *Federal Register* notice on February 3, 1997, (62 FR 4941). A second 30-day extension was announced in a third *Federal Register* notice on March 20, 1997 (62 FR 13355). This notice extended the public comment period to April 16, 1997.

Forecast: A draft final Notice of Proposed Rulemaking will be prepared after the DOE has considered the public comments. The draft final version will then be submitted to the NRC for their concurrence review.

2.2.2 Environmental Compliance Activities

During the reporting period, several environmental surveys and permitting actions were completed for new and planned site characterization activities at Yucca Mountain. Progress in these areas is reported in this section.

2.2.2.1 Permits

Permits are required for some land use activities having potential environmental impacts. This section discusses progress or activity associated with these permits.

Water Quality

Quarterly bacteriological sampling of the Exploratory Studies Facility (ESF) potable water system, required by the Nevada Department of Health, continued this reporting period. The state-certified laboratory that analyzed the samples reported the absence of coliform bacteria in the system. In addition, as required by the septic tank general discharge permit, the annual discharge monitoring report, filed in January 1997 for the ESF septic/leachfield system, documented compliance with permit conditions.

Two waiver time extensions for permits 58827 and 58829 were submitted to the State Engineer for continued testing at the boreholes at the C-hole complex.

Forecast: Submittal of water quality permits will continue in FY 1997, as needed.

Air Quality

On November 25, 1996, a revised Air Quality Operating Permit No. AP9611-0573 was received. The revised permit included numerous clerical corrections and the following three changes requested by the Project: (1) the deletion of the grout batch plant, the CME 85 drill rig, and two Top Head Drive Core drill rigs; (2) the reduction of annual operation hours for the LM300 drill rig; and (3) the reduction of annual operation hours for the Atlas Copco air compressors. The reduction in hours will help maintain the threshold for the Class II Air Quality Operating Permit.

Forecast: Because of changing Project needs, a change in operational hours for certain permitted air emission sources is projected in the second half of FY 1997. The submission of new air quality operating permit applications will continue in FY 1997, as needed.

Underground Injection - Drilling and Tracers

Two tracer tests were requested for the C-hole complex. The tests were conducted to evaluate hydrogeologic properties of the Bullfrog Formation. Tracers currently approved under the Underground Injection Control Permit UNEV89031 were used. Approval was granted by the State of Nevada for the following tests:

1. Injection of lithium bromide, pentafluorobenzoic acid, and fluorescent microspheres in borehole UE-25 c#2 at C-hole complex. Approval was granted on October 8. 1996.

2. Concurrent injection of pyridone into borehole UE-25 c#1 and difluorobenzoic acid into borehole UE-25 c#2 and at C-hole complex. Approval was granted on December 16, 1996.

Approval from the State of Nevada was granted on October 11, 1996, for a request to install an inflatable borehole liner at borehole USW UZ-14. The liner is a temporary instrument capable of providing ambient pneumatic data. The liner was installed in November 1996.

On November 26, 1996, the second ventilation tracer test for the ESF ventilation system was requested. The second test was to evaluate leaks at both the tunnel boring machine cutter head and the trombone air inlet area. The use of 3.75 cubic feet of sulfur hexafluoride tracer was requested, and approval was granted on December 16, 1996. The test was conducted on February 26, 1997.

A letter proposing the use of oil skimmer-treated water for dust control was submitted to the State of Nevada on January 29, 1997. Approval is expected in early April.

Forecast: Tracer requests for drilling new boreholes into the water table and conducting pump tests are expected in the second half of FY 1997. Submittal of additional tracer test requests for the C-hole complex and the ESF ventilation system will continue in the second half of FY 1997, as needed. A request to modify the underground injection control permit to include new reactive tracers proposed for C-hole testing is also projected in the second half of FY 1997.

2.2.2.2 Environmental, Safety, and Health Assessments

Comprehensive and focused, special-issue environmental, safety, and health assessments are performed under a Project proactive assessment program to evaluate organizational and programmatic compliance with Federal and State statutory requirements, DOE Orders, and Project plans and procedures. Comprehensive assessments evaluate a broad range of environmental, safety, and health topics in a specific organization or Project-wide program.

Focused special-issue assessments examine a particular program element, a potential noncompliance condition, or a specific area of concern that is time-sensitive or requires immediate attention.

Comprehensive assessments during this period evaluated the Radiation Protection Program and the Permit Compliance Program. Both programs were found to be in compliance with external regulatory and Project-directed requirements. Recommendations were provided to improve those program elements and practices relating primarily to administrative and training requirements. Focused assessments performed during this period evaluat management of hazardous waste satellite accumulation areas, used oil, and heavy equipme elative to applicable environmental, safety, and health requirements. As an overall result of these assessments, specific program elements were revised and implemented to improve management practices in these areas. Another assessment examined trends from all past assessments and identified general areas where opportunities exist to enhance specific environmental, safety, and health program elements.

Forecast: Two comprehensive Environmental, Safety, and Health Assessments and four focused, special-issue assessments are planned for the period April 1, 1997, through September 1997.

2.2.2.3 Environmental Surveillance

Approximately 140 environmental surveillances were conducted at the Yucca Mountain site to ensure compliance with environmental, programmatic, and permit requirements. Corrective action and follow-up work were required on 17 (12 percent) of the surveillances, and 3 (about 20 percent) of those follow up activities were completed during this reporting period.

Forecast: Approximately 170 surveillances are projected for the second half of FY 1997. Remaining corrective action and follow-up work from the first half of FY 1997 will be completed during the second half of FY 1997.

2.2.2.4 Preactivity Surveys

During this reporting period, three land access and environmental compliance activity reviews were completed, and one partially completed preactivity survey for the south portal was canceled because of a requirements change. The preactivity process involves acquiring land access approvals and right-of-way reservations and completing environmental preactivity surveys (which include archaeological, biological and, in certain instances, radiological surveys) before any Project site activity can be initiated. During the first half of FY 1997, one request to initiate preactivity surveys was received.

Forecast: Approximately three to six land access and environmental compliance activity reviews are expected to be completed during the second half of FY 1997

2.2.2.5 Land Acquisition and Site Access

One new participant request to initiate a site characterization activity was received during this reporting period. The request (Installation of Seismic Stations by Los Alamos National Laboratory) was completed and access authorization granted.

Three requests from the last reporting period have not yet been completed. These include two requests (Installation of University of Nevada, Reno, Seismological Laboratory Seismic Instrumentation in Y-Tunnel and Post Activity Survey for Four Seismic Stations) that are awaiting Nevada Test Site Operations Permits. The third request (Amargosa Desert Digital Seismic Station for Southern Great Basin Seismic Network) is awaiting a right-of-way reservation from the Bureau of Land Management.

Forecast: Three to six land access and environmental compliance activity reviews are expected to be completed during the last half of FY 1997.

2.2.3 Licensing Activities

The DOE's eventual goal is to obtain the necessary licenses and permits for the repository, if the site is found suitable. In pursuit of this goal, licensing activities included the management of and participation in interactions with the NRC and other oversight organizations. These interactions help to clarify regulatory and technical issues and to reach a common understanding of regulatory requirements.

2.2.3.1 Interactions with the U.S. Nuclear Regulatory Commission and Other Organizations

This section reviews and discusses significant actions, agreements, and accomplishments that resulted from interactions between DOE and NRC, meetings of the Advisory Committee on Nuclear Waste, and meetings of the Nuclear Waste Technical Review Board. Appendix D tabulates the interactions with each of the following agencies.

U.S. Nuclear Regulatory Commission

From October 1996 through March 1997, the DOE participated in several interactions with the NRC, including one technical exchange on the igneous activity program. In addition there were two technical meetings (on ESF construction, scientific studies and testing and design status) and two management meetings. Numerous informal interactions also occurred between DOE personnel and NRC onsite representatives, including regularly scheduled meetings with the Yucca Mountain Site Characterization Office (YMSCO) Project Manager and Assistant Managers. Also, three Appendix 7 meetings, one on Disposal Criticality Analysis Methodology, one on Level of Design Detail, and one on Seismic Topical Report II (DOE, 1996c) tectonics models were held during this period. Appendix 7 meetings are informal meetings conducted under the procedures outlined in Appendix 7 of a procedural agreement between NRC and DOE on the principles for interface during geologic site investigation and site characterization (DOE, 1993a). Appendix 7 deals specifically with NRC onsite representatives.

The DOE and the NRC hold periodic management meetings to provide a forum for management-level discussions of issues and concerns associated with the Yucca Mountain Site Characterization Project and other aspects of the CRWMS program. Two such meetings were held on October 23, 1996, and January 15, 1997, by video conference among multiple sites. The locations involved were YMSCO, DOE Headquarters, NRC Headquarters, and the Center for Nuclear Waste Regulatory Analyses in San Antonio, Texas. These meetings covered a number of topics, including an update of the Program Plan and budget, the legislative process, an update of Office of Waste Acceptance and Storage and Transportation activities, and regulatory and licensing issues and topics. The regulatory and licensing topics discussed included the status of 10 CFR Part 960, an update on DOE documentation of decisions, the development of the third seismic topical report, an update of the licensing support system, and NRC QA concerns. Also, an overview was presented of the NRC High-Level Radioactive Waste Program Annual Progress Report Fiscal Year 1996 (NRC, 1997a) before its distribution.

On February 25-26, 1997, at the NRC offices in Rockville, Maryland, the DOE and the NRC conducted a technical exchange on igneous activity. The purposes of the exchange were (a) to define the approach for considering the igneous activity issue in the total system performance assessment supporting the viability assessment and (b) to identify areas of agreement and disagreement on the relevant geologic data, the probability of volcanism, models for calculating consequences, and performance assessment models of igneous activity. Topics discussed at the exchange included the geologic setting and relevant data from the NRC, field studies of the Center for Nuclear Waste Regulatory Analyses, ground magnetic surveys in the Yucca Mountain region, definition of the Yucca Mountain regional system, results from DOE's probabilistic volcanic hazard assessment, DOE plans for the total system performance assessment supporting the viability assessment, NRC staff probability models and their concerns with source zone definitions for the probabilistic volcanic hazard analysis, the structural setting of the Yucca Mountain region relevant to the repository, and integrated volcanism structural models. In addition, the NRC staff presented consequence models for tephra dispersion, subsurface area of dispersion and critical models used in performance assessment, and sensitivity studies performed by the Center for Nuclear Waste Regulatory Analyses.

The technical exchange was significant in that both the NRC staff and the DOE agreed that the issue of igneous activity can be resolved by defining a path for resolution and by identifying the areas where additional work may be required. In closing, both parties further discussed the points of agreement between the NRC and DOE and the proposed path to resolve the subissue of probability of igneous activity. The following agreements reached by the NRC and the DOE are believed to provide a path forward for the resolution of the igneous activity issue:

1. DOE and NRC agree that (a) the rate of volcanism is relatively constant for the last 5 million years and can be assumed to remain relatively constant for the period of

performance; (b) current information indicates silicic volcanism need not be further evaluated; (c) volcanism is of regulatory interest and its probability and consequences will be considered; (d) if determined to be significant with respect to repository performance, the effects of volcanism will be included in the total system performance assessment; (e) the treatment of consequences outlined by DOE that includes extrusive magmatic events (cone and dike formation) and intrusive magmatic events (sill and dike formation) with both direct and indirect effects is generally appropriate at the level of detail provided; and (f) there is uncertainty in consequence analysis for magmatic waste package interactions and that DOE will evaluate this uncertainty.

2. DOE agrees to consider evaluating, through hazard sensitivity studies, new data such as the following:

   - The size and volume of Little Cone

   - The number of events at Anomaly A (a possible buried cinder cone, or intrusion, south of Lathrop Wells cone).

3. NRC believes that a probability of 1×10^{-7} per year is a reasonably conservative upper bound for extrusive events. There are differing views on the lower bound. DOE considers that the probabilistic volcanic hazards assessment provides a defensible basis for characterizing the probability of disruption (includes both intrusive and extrusive magmatic events). The probability distribution function has an upper bound frequency of 1×10^{-7}, a lower bound of 1×10^{-10}, and a mean of 1.5×10^{-8} per year. DOE agrees to explain how the probability distribution function for the probability of disruption will be used in performance assessment, including sensitivity studies, recognizing NRC's comments.

4. DOE agrees to provide the NRC with a letter describing the DOE basis for subissue resolution, as specified in 2 and 3, for consideration in the development of NRC's Issue Resolution Report

The DOE and the NRC staff conducted technical meetings on December 16, 1996, and March 13, 1997, to discuss the status of ESF construction, design, and ESF-related site characterization activities, and to resolve identified issues. Participants in the multisite video conference were from the NRC offices in Rockville, Maryland, the DOE headquarters, YMSCO, and the Center for Nuclear Waste Regulatory Analyses in San Antonio, Texas. Major topics addressed during the meetings were related to the status of the ESF tunnel and alcove construction, scientific studies and testing update, and description of the engineering design program. Also discussed during the December 16, 1997, technical meeting were NRC staff concerns related to alcove excavation methods, drill-and-blast and mechanical excavation, and thermal tests identified during a July 24, 1996, Appendix 7 meeting on thermal tests. During the March 13, 1997, DOE presented its plans for retrievability. Also discussed were the potential of including in future quarterly meetings a broad range of technical topics related to the ESF and the proposed repository construction, testing, design, and performance assessment.

In addition to the formal interactions with the NRC staff, regularly scheduled informal interactions with the NRC onsite representatives and Appendix 7 meetings were conducted to discuss and maintain communication with the NRC staff on technical and programmatic matters of interest to both agencies. On February 5, 1997, an Appendix 7 meeting on disposal criticality analysis methodology was held in Washington, D.C. Participants were from OCRWM, YMSCO, and the NRC Division of Waste Management. The DOE's objectives for the meeting were (a) to discuss the DOE's proposed postclosure disposal criticality analysis methodology, (b) to understand and address NRC staff concerns and questions on the methodology and on the Disposal Criticality Analysis Methodology Technical Report (CRWMS M&O, 1996d), (c) to better understand the NRC staff's views regarding identified disposal criticality rule issues, and (d) to seek NRC staff feedback on the likelihood of acceptance of the planned criticality analysis methodology. The NRC's objective was to provide DOE early feedback on the acceptability of its proposed disposal criticality analysis methodology. At the conclusion of the meeting the DOE recognized the NRC's concerns and comments, and noted that additional work will be needed to justify the approach in support of a potential license application.

An Appendix 7 meeting on design detail for the license application was held on February 6, 1997, at the NRC Headquarters, in Rockville, Maryland. Participants included OCRWM, YMSCO, and the NRC Division of Waste Management. The DOE's objectives for the meeting were to update the NRC participants on the repository design, to provide an overview of DOE's approach to determining and developing the appropriate level of design detail for the license application, to provide examples of application of that approach, and to seek NRC feedback on the approach.

An Appendix 7 meeting between the NRC and DOE staffs was held on February 27, 1997, to discuss resolution of NRC staff comments on the second in the series of DOE seismic topical reports entitled Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain (DOE, 1996c). The meeting discussions indicated that all NRC staff comments could be resolved with relatively minor changes and clarifications to the topical report. The DOE expects to receive an NRC staff letter to this effect and plans to revise and reissue the topical report accordingly.

Advisory Committee on Nuclear Waste

The DOE's participation in the NRC's Advisory Committee on Nuclear Waste meetings was limited during this period. Some of the meetings attended by the DOE staff included discussions of topics related to the OCRWM and the Yucca Mountain Site Characterization Project. The following paragraphs provide the highlights of the 87th, 88th, 89th, and 90th Committee meetings.

The 87th meeting of the Advisory Committee on Nuclear Waste meeting was held October 22-23, 1996 in Rockville, Maryland. This meeting mostly dealt with the administrative topics, future Committee activities and plans, and the preparation of the Committee reports. One topic of interest to the Project discussed was the Branch Technical Position on Requirements for

Radioactive Waste Land Burial Sites Authorized Under Former 10 CFR 20.302, 20.304 and current 20.2002. This meeting was not attended by Project personnel.

The 88th meeting of the Advisory Committee on Nuclear Waste was held November 12-13, 1996, in Rockville, Maryland. The meeting was attended by the DOE-HQ staff. The meeting was originally planned as a retreat and thus, was primarily concerned with planning and issues internal to the Committee. Discussed were the conduct of Committee activities, internal operations and methods for formulating advice, and priority issues for Committee consideration. The Committee also discussed the preparation of proposed Committee reports and future Committee activities and agendas.

The 89th meeting of the Advisory Committee on Nuclear Waste held January 28-30, 1997, in Rockville, Maryland. Reports discussed that were relevant to Yucca Mountain included the Radionuclide Transport at Yucca Mountain, Critical Group and Reference Biosphere for a Waste Disposal Facility Performance Assessment, and Time of Compliance in Low-Level-Waste Disposal. Also discussed were the status of site characterization at the proposed Yucca Mountain repository, the status of the NRC staff's efforts to revise 10 CFR Part 60, the status of an NRC staff paper giving options for NRC response to the DOE-proposed revision of 10 CFR Part 960, and the status of U.S. Environmental Protection Agency rulemaking activities on a new public health and safety standard for Yucca Mountain. This meeting was attended by Project personnel, however, they did not make any presentations.

The 90th Advisory Committee on Nuclear Waste meeting, conducted March 20-21, 1997, in Rockville, Maryland, was attended by Project staff. Presentations were given on the status of the NRC staff review of DOE's siting guidelines (10 CFR Part 960) and on Phase II of the Biosphere Model Validation Study (BIOMOVS II). A representative from the Electric Power Research Institute discussed the biosphere modeling and dose assessment for Yucca Mountain; this modeling is associated with the BIOMASS theme sponsored by the International Atomic Energy Agency. A representative of the Division of Waste Management discussed the major activities the Division will focus on in FY 1997 and proposed a number of interactions with the Committee to discuss those activities. Also discussed were the historical perspective of defense in depth philosophy and the Committee letter to the Commission on the biosphere and critical group considerations for Yucca Mountain.

Nuclear Waste Technical Review Board

During the last six months, the Nuclear Waste Technical Review Board held two meetings to discuss issues related to the OCRWM Program. The full Board meetings provided a forum for discussion of issues related to the management and disposal of high-level radioactive waste between the Board, program participants, representatives of State and Federal agencies, and the public. At the October 9-10, 1996, meeting in Arlington, Virginia, Dr. Daniel Dreyfus, OCRWM Director, presented an overview of the viability assessment, highlighting its significance to the Program and its relevance to the nation's radioactive waste management policy. In a series of presentations, Project staff discussed aspects of the Program related to repository design, status of exploration and testing, and detailed plans for the components of the viability assessment

During the meeting, several Board members commented favorably on the substantial progress the Program had made in 1996 and the evidence of better integration of Project activities. In addition, several Board members commented on the need for the Program to retain flexibility in the repository design to accommodate new information resulting from the viability assessment.

During its January 28-29, 1997, meeting in Pahrump, Nevada, the Board met to discuss such topics as transportation, total system performance assessment, Program and Project activities and investigations, and reduction of hydrologic uncertainties. Following Project staff presentations, a round-table discus on addressed transportation-related topics. Participants included the DOE, the State, and local and public interest groups. OCRWM updated the status of activities related to the upcoming viability assessment, the privatization initiative for transportation services, and the generic planning and analysis for possible interim storage of spent nuclear fuel. Hydrogeologic modeling efforts in both the saturated and unsaturated zones at Yucca Mountain, thermal and underground testing, and the scientific work being conducted by Nye County, Nevada, were also discussed on the second day.

On October 3, 1996, the Department responded to recommendations made by the Board in its 13th report entitled "Report to the U.S. Congress and the Secretary of Energy— 1995 Findings and Recommendations" (NWTRB, 1996). In its report, the Board stated that OCRWM had made significant progress in characterizing the Yucca Mountain site, despite the budgetary, programmatic, and regulatory uncertainties facing the Program, but it made numerous technical recommendations for strengthening the site characterization program. The Board specifically highlighted the need for adequate and stable funding for the Program to achieve its objectives.

On March 31, 1997, the Board issued its 14th report entitled "Report to the U.S. Congress and the Secretary of Energy— 1996 Findings and Recommendations" (NWTRB, 1997). In this report, the Board summarized the major findings, conclusions, and recommendations that have resulted from Board activities during calendar year 1996. While recognizing that much scientific progress has occurred at Yucca Mountain, the Board identified three major areas of concern. These concerns involved the distinction between the viability assessment and site suitability, the need for excavation of an east-west tunnel to determine site suitability, and the evaluation of alternative repository designs. In addition, the Board identified several additional enhancements to the Program and recommended that (1) [illegible] dvantages of the multi-purpose canister program, such as standardization, be incorporate[illegible] development of the market-driven approach for waste acceptance and transportation serv[illegible] and (2) the total system performance assessment be "transparent" (easily understood), "valid" (reasonably accurate and representative of actual conditions), properly treat uncertainty, and objectively peer reviewed. The Board also recommended that public understanding and acceptance of total system performance assessment be increased through broader public involvement in the process. The report noted that developing the waste isolation strategy slowed in 1996 because of difficulty in reaching consensus on the issues. The Board, however, was "satisfied that the goal of articulating a clear waste isolation strategy seems to be serving its purpose."

U.S. Environmental Protection Agency

Since the last progress report, the status of a new public health and safety standard for Yucca Mountain, as well as the DOE's view on the issues that are important to the ability to implement the standard, have not changed.

Forecast: The next key step in this process is the interagency review of the draft rule that is coordinated by the Office of Management and Budget. Regular technical interactions and exchanges with the NRC will continue, as will meetings with the Advisory Committee on Nuclear Waste and Nuclear Waste Technical Review Board.

2.2.3.2 Issue Resolution

The DOE's approach to licensing and issue resolution has evolved from the process outlined in Section 8.1 of the Site Characterization Plan (SCP) (DOE, 1988), which focused on resolving individual issues related to specific site characteristics. Issues were dealt with in relative isolation from one another and from an overall concept for the repository system. Experience gained in applying this process, however, indicates that the sufficiency of site characterization data and analyses can generally only be determined within the context of a coherent repository concept that includes both a design and an assessment of its performance in the geologic setting. The DOE believes that licensing success will depend in part on establishing such a conceptual framework and on focusing its near-term interactions with the NRC within the context of this framework.

The DOE is currently in the process of developing an overall repository concept to support the viability assessment and will communicate its progress to the NRC. The concept is expected to evolve over time as the design, site data, and performance analyses mature. As this conceptual framework develops, the DOE will seek to reach a common understanding with the NRC regarding the issues that are significant to the overall performance of a repository at Yucca Mountain. The DOE also will seek to reach agreement on the methodologies and approaches used to address important technical issues such as criticality control and seismic design. A limited number of topical reports will be developed with the goal of receiving NRC safety evaluation reports that can be referenced in a license application as an appropriate means for resolving selected issues.

The goal of this new approach is to reach a mutual understanding of the developing repository concept that will provide a basis for the NRC's preliminary comments on the sufficiency of the DOE's site characterization analysis and design for inclusion in a license application. The revised approach is consistent with the intent of the issues-based approach that was described in the SCP. This approach also reflects the substantial change in the policy framework for repository development and the increased understanding of the Yucca Mountain site and repository design that have occurred since the SCP was written

Activities conducted during the reporting period that relate to the evaluation of seismic hazards, igneous activity, and repository criticality are summarized in the following sections. The status of actions related to open items from the NRC's Site Characterization Analysis (NRC, 1989) is also discussed.

Seismic Hazards

The DOE revised the topical report, Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain (DOE, 1994b) to reflect the resolution of NRC staff comments and to incorporate the results of recent work. The resolution of staff comments is documented in an NRC staff letter dated July 25, 1996 (NRC, 1996b). The revised topical report is undergoing QA review, and the DOE plans to issue Revision 1 of the report during the next reporting period.

An Appendix 7 meeting between NRC and DOE staffs was held on February 27, 1997, to discuss resolution of NRC staff comments on the second in the series of DOE seismic topical reports, Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain, Rev. 1 (DOE, 1996c). The meeting discussions indicated that all NRC staff comments could be resolved with relatively minor changes and clarifications to the topical report. The DOE expects to receive an NRC staff letter to this effect and plans to revise and reissue the topical report, accordingly.

The DOE plans to prepare the third and final seismic topical report during FY 1998. This report will present the preclosure seismic design inputs for repository facilities that are important to safety, based on the results of the application of the methodologies in the first two topical reports. After the third seismic topical report has been issued, the DOE expects that the NRC staff will prepare a Prelicensing Evaluation Report for the DOE's seismic hazards evaluation and preclosure seismic design methodologies and the preclosure seismic design inputs. A Prelicensing Evaluation Report would be similar to a Safety Evaluation Report; however, no licensing precedent exists for a Prelicensing Evaluation Report.

Igneous Activity

Extensive volcanism studies and data collection for the DOE Yucca Mountain Site Characterization Project have been conducted since 1979 to provide a scientific basis for volcanic hazard assessment and to assist in applying the data to the regulatory requirements for siting a potential repository at Yucca Mountain. Igneous activity has been identified by the NRC staff as a key technical issue for the Yucca Mountain site as part of the issue resolution activities on the subject of volcanic hazards.

The DOE igneous activity program is directed toward evaluating the significance of igneous activity by reviewing and independently confirming the data, and evaluating and developing alternative conceptual models for the probability and consequences of igneous activity at Yucca Mountain. Reasonable bounding ranges of probability and consequences of igneous activity will be used to assess the impact on repository performance. During the previous reporting period, the Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada, was submitted to the NRC on June 25, 1996 (CRWMS M&O, 1996e). The report documents the results of an expert elicitation to assess the probability of disruption of the potential high-level waste repository at Yucca Mountain, Nevada, by igneous events and quantifies the uncertainties associated with this assessment.

Since the last submittal of this report, there have been two interactions with the NRC staff to discuss paths for resolution of the igneous activity issue. During an Appendix 7 meeting on September 10, 1996, the NRC staff provided feedback and comments on DOE's Probabilistic Volcanic Hazard Analysis report. The NRC staff's comments mostly dealt with the elicitation process, documentation of the results, and utilization of new data, as compared to the guidance provided in the NRC Branch Technical Position on expert judgment (NRC, 1996a). During this meeting the NRC concluded that the Probabilistic Volcanic Hazard Analysis generally appeared to be consistent with the Branch Technical Position and provided some specific recommendations to consider for subsequent expert elicitation projects. The NRC also indicated that, as a result of the Probabilistic Volcanic Hazard Analysis, the NRC's Site Characterization Open Item 3 has been addressed and several other comments related to the Study Plan 8.3.1.8.1.1 will be closed by a separate NRC staff letter to DOE. The NRC staff provided guidance for closure of Comment 3 regarding reliance on expert judgment for licensing and Comment 7 regarding use of expert judgment versus peer review in a letter to DOE on December 26, 1996 (NRC, 1996c).

On February 25-26, 1997, a technical exchange was conducted in the NRC offices in Rockville, Maryland. The purpose of the exchange was to define the approach to considering igneous activity in total system performance assessment for the viability assessment and to identify areas of agreement and disagreement on the relevant geologic data, the probability of volcanism, models for calculating consequences, and performance assessment models of igneous activity. Both NRC and DOE staffs made presentations.

At the conclusion of this meeting, both agencies developed a list of agreements and defined additional work needed to resolve the igneous activity issue. A summary of this interaction and a complete list of DOE and NRC agreements in defining a path for resolution of the igneous activity issue are provided in Section 2.2.3.1 of this progress report.

Repository Criticality

The disposal criticality issue involves demonstrating that criticality control will be maintained in the repository such that 10 CFR Part 60 criticality requirements and repository performance objectives will be met. Fissile material remains in spent nuclear fuel after discharge from a reactor, and much of that material is very long lived. Though this material gradually decays to nonfissile materials, additional fissile materials resulting from the decay of some nonfissile materials are created after the spent fuel is emplaced in the repository. Therefore, disposal criticality analysis is required for the period from the time of emplacement to thousands of years in the future.

The Project's approach to dealing with the issue of disposal criticality involves a risk-based criticality analysis methodology that will be used to demonstrate that potential criticality events during the postclosure period will not pose an unacceptable risk to the health and safety of the public. This approach is consistent with the 1995 recommendations of the National Academy of Sciences for Yucca Mountain standards (NAS, 1995).

The Project will also seek NRC approval of accounting for the reduced reactivity associated with spent nuclear fuel as compared with unirradiated fuel of the same initial enrichment. This approach is known as bur[illegible]edit. Additional possible criticality control methods are being considered, including the [illegible]tion of supplemental neutron absorbing materials and/or moderator-displacing materials to the waste package.

Resolution of disposal criticality issues will be based primarily on the Disposal Criticality Analysis Methodology Topical Report, which is planned for submittal to the NRC in 1998. This topical report will seek NRC acceptance of the DOE's proposed criticality analysis methodology and its method for obtaining burnup credit.

During the previous reporting period, the DOE sent the Disposal Criticality Analysis Methodology Technical Report (CRWMS M&O, 1996d) to the NRC staff for comment and feedback. This report provided a preliminary description of the DOE's proposed risk-based disposal criticality analysis methodology. It will be used as the basis for developing the Disposal Criticality Analysis Methodology Topical Report. During this reporting period, DOE received NRC staff comments on the technical report. On February 5, 1997, the DOE and the NRC held an Appendix 7 meeting in Washington, D.C., to discuss comments from NRC staff and to begin work toward resolving issues identified by the comments.

Forecast: Additional DOE-NRC meetings may occur in upcoming reporting periods to help resolve open issues. During the next reporting period, the DOE will continue to develop the disposal criticality analysis methodology and its supporting models. This effort is planned to lead to a revision to the Disposal Criticality Analysis Methodology Technical Report, scheduled for completion in late FY 1997. DOE plans to reissue the second seismic topical report and to proceed on work in accordance with the agreements on igneous activity established in the Appendix 7 meeting held in February 1997.

Resolution of Site Characterization Analysis Open Items

The NRC has provided guidance for closure of Site Characterization Analysis Comments 3 and 7 relating to Use of Expert Judgment (NRC, 1996c).

At the end of this reporting period, 98 Site Characterization Analysis open items have been closed by the NRC (including 2 objections), and 100 items remain open. Most of the remaining open items await data to be acquired through site characterization activities for resolution. Of the remaining 100 Site Characterization Analysis open items, 32 are currently being reviewed by the NRC staff.

Supplemental responses to Site Characterization Analysis Comment 24 (DOE, 1996d) and Comment 31 (DOE, 1996e) were forwarded to the NRC for consideration.

Appendix E provides the status of Site Characterization Analysis comment resolution, identifying actions that need to be performed to close the item.

Forecast: The Project will continue discussions with NRC to address questions and concerns that may result from the staff's continued review of Seismic Topical Reports I and II.

Activities related to disposal criticality during the next reporting period will be focused on further development of the Disposal Criticality Analysis Methodology Topical Report and on continuing discussions with the NRC regarding the criticality rule in 10 CFR 60.131(b)(7).

Site Characterization Analysis open items will continue to be resolved as the site characterization and other programmatic activities provide pertinent data.

2.2.3.3 License Application

As discussed in Progress Report #15, this section (2.2.3.3) will be titled "License Application" from this progress report forward and will report progress in planning for and development of the license application. The license application is planned to be submitted to the NRC in 2002 to request authorization to begin repository construction. Submittal of the license application would follow successful accomplishment of activities that, by law, precede it. These activities include the following:

- Completion of the repository environmental impact statement
- DOE recommendation of the site to the President
- Presidential recommendation of the site to the Congress
- The site recommendation becoming effective as provided for in the Nuclear Waste Policy Act of 1982.

During this reporting period, the DOE began developing three products that will support development of the license application. These products are the License Application Plan, the License Application Management Plan, and the Technical Guidance Document for Preparation of the License Application.

The License Application Plan will describe the Project work needed to produce a docketable license application between the viability assessment and the license application submittal, give a schedule for this work, and provide a cost estimate for this work. The License Application Plan will also outline the Project's licensing strategy and give the content requirements for the license application. This plan is one of the four products that will be prepared to support the viability assessment. The information developed for the License Application Plan will also be used to support the development of the license application, if the site is determined to be suitable.

The License Application Management Plan will provide the management framework within which the license application will be developed. Information included will be the layout of the license application, the document control process for the license application, and a step-by-step description of the process by which the license application will be developed.

The Technical Guidance Document for Preparation of the License Application will eventually provide guidance to authors on the technical and licensing content of their license application chapters. The technical guidance document will list the regulatory requirements applicable to each chapter and will provide acceptance criteria for the authors to use to verify they have provided an adequate licensing case in their chapters. The document will also describe how industry standards and other available guidance should be used in the licensing case to be presented in the license application.

In addition, the Management Plan was developed for the Project Integrated Safety Assessment (DOE, 1997a). Project staff then began drafting various sections of the Project Integrated Safety Assessment, which will be used as a starting point for developing the license application. The document will integrate current knowledge of the site obtained from site programs, design, and performance assessment and contain information in a format suitable for updating into a Safety Analysis Report. The document may also be used to initiate the NRC's preliminary sufficiency comments as required by the Nuclear Waste Policy Act. The Project Integrated Safety Assessment is scheduled to be completed in August 1998.

Forecast: The Draft License Application Plan and the Draft Technical Guidance Document, both based on information available in FY 1997, will be completed by the end of September 1997. Revisions to both documents are planned for FY 1998 to incorporate additional information expected to be available at that time. The License Application Management Plan is expected to be completed by the end of FY 1997.

2.3 QUALITY ASSURANCE

The DOE is continuing to monitor the status and adequacy of the OCRWM QA Program. In addition, the DOE continues with the reengineering of the OCRWM QA Program started during the Progress Report #12 reporting period. The reengineering effort began with DOE initiatives to improve effectiveness and to cut costs.

2.3.1 Program Activity

Revision 6 of the Quality Assurance Requirements and Description document was issued during this period (DOE, 1997b). This document is the principal QA requirements document for the Civilian Radioactive Waste Management Program and contains regulatory requirements and commitments necessary to develop an effective QA program. Revision 6 was developed to address and incorporate NRC comments concerning scientific investigation and design control.

Ten internal audits were conducted during this reporting period. Audits are verification activities that are planned and documented to evaluate compliance with requirements and to determine effectiveness of implementation through a review of objective evidence. The following organizations were audited: U.S. Geological Survey, Kiewit/PB, DOE Office of Environmental Management and the CRWMS M&O, including its activities at national laboratories.

Forty-five surveillances were also conducted during this reporting period. Surveillances are observations of real-time quality-affecting activities and/or the review of documentation to verify conformance with specified requirements. The surveillances evaluated specific site characterization activities, including the adequacy and effectiveness of corrective actions taken to resolve previously reported conditions.

Results of audits and surveillances conducted this reporting period indicated that the OCRWM QA Program was in compliance with requirements, effectively implemented, and satisfactory overall. None of the verification activities conducted this reporting period identified conditions that warranted initiation of immediate corrective action or issuance of a stop work order.

In addition to the internal audits and surveillances, a total of 17 supplier audits and 1 supplier qualification survey were performed. Deficiencies identified during these activities are being monitored by the DOE to ensure satisfactory disposition.

During the first half of FY 1995, the DOE initiated a plan to reengineer the QA function on the OCRWM Program. The first phase of this plan, titled "Reengineering of the Quality Assurance Function on the Civilian Radioactive Waste Management Program Transition Plan" (DOE, 1994c) was completed during the Progress Report #13 reporting period. This phase transferred certain verification functions (that is, internal audits, supplier audits and supplier qualification surveys) performed by the individual affected organizations to the OCRWM Office of Quality Assurance. In addition, the DOE assumed complete responsibility for tracking and trending Corrective Action and Nonconformance Reports.

During this reporting period, DOE completed the initial part of Phase II of the ongoing reengineering of the QA function on the OCRWM Program. The implementation of this portion of the reengineering plan was controlled by the plan "Reengineering the Quality Assurance Function on the Office of Civilian Radioactive Waste Management Program Consolidation of OCRWM QA Functions, Phase II Transition Plan, Phase A" (DOE, 1997c). This phase transferred the surveillance function performed by the individual affected organizations to the OCRWM Office of Quality Assurance. The effective date for Phase A implementation was February 1, 1997.

Forecast: The plan to control the implementation of the final phase of the reengineering of the QA function within the OCRWM Program is currently in the approval process. Once approved and implemented, this plan will transition the remaining QA functions performed by the individual affected organizations to the OCRWM Office of Quality Assurance. The Office of Quality Assurance will then have complete responsibility for the management and performance of the QA function within the OCRWM Program. Implementation of the final phase of the transition is scheduled to occur in the next reporting period.

2.3.2 Determination of Importance Evaluations

The DOE is responsible for performing determination of importance evaluations, which remain the mechanism established by the Project to evaluate proposed field activities with respect to their potential for adverse impact to Q-List items or site characterization testing. Because the natural barriers are included in the Q-List, each determination of importance evaluation essentially comprises both a waste isolation evaluation and a test interference evaluation for the proposed activity. Where a reasonable potential for adverse impact is identified, the evaluation establishes appropriate QA controls for the activity to prevent or limit the adverse impact. These controls are then transcribed into the applicable documents for implementing the activity (for example, field work packages, design packages, specifications, and drawings).

The implementing line procedure NLP-2-0, *Determination of Importance Evaluations* that controls this process requires that every determination of importance evaluation be reviewed annually to determine whether critical inputs or assumptions have changed enough that the conclusions of the earlier determination of importance evaluations may no longer be valid.

During the reporting period, a number of determination of importance evaluations received their annual review as required by procedure. No significant changes were identified in either critical inputs or assumptions for the reviewed determination of importance evaluations. Various minor changes were recommended to improve consistency and to update references.

In addition, a number of tracers, fluids, and materials reports from various responsible organizations were reviewed before inclusion in the tracers, fluids, and materials data base. A list of determination of importance evaluations performed during this reporting period is provided in Appendix F.

Forecast: Continuing determination of importance evaluation development work is scheduled for the next reporting period, predominantly in support of surface-based testing and testing in the subsurface ESF.

2.3.3 Q-List and Management Control List/Design Basis Event Analysis

The Q-List (DOE, 1997d) and Management Control List (DOE, 1994d) delineate permanent items (that is, items that may become part of a licensed pre- or postclosure repository). The Q-List includes those permanent items determined to be important to radiological safety, important to waste isolation, or otherwise subject to the requirements of the Quality Assurance Requirements and Description document (DOE, 1997b). The Management Control List includes those permanent items determined not to be subject to those requirements; they are instead subject to conventional quality management and design controls.

Both documents reflect the conclusions of classification analyses. The Q-List also contains items originally placed on it by "direct inclusion," a conservative approach initiated when the document was first issued. Such items can only be removed through documented analysis.

Classification analyses support the Implementation Line Procedure that identifies whether a permanent item has a function that falls within one of the following seven categories (order does not imply relative importance):

QA-1 Important to Radiological Safety

QA-2 Important to Waste Isolation

QA-3 Important to Radioactive Waste Control

QA-4 Important to Fire Protection (applicable to QA-1 or -2 items)

QA-5 Important to Potential Interaction (this does not include permanent item function, but, rather, failure impact to QA-1 or -2 items)

QA-6 Important to Physical Protection of Facility and Materials

QA-7 Important to Occupational Radiological Exposure.

The classification analysis for ground support systems was revised during the reporting period. The revision clarified ESF permanent function ground support maintenance requirements. Conclusions in this evaluation revision are consistent with the previous ground support evaluations. A classification analysis was also performed for the offsite transportation configuration item. (Offsite transportation had conservatively been included on the Q-List by direction.) The analysis concluded that offsite transportation did not meet any of the QA classification categories and could therefore be removed from the Q-List.

The DOE approved and issued a revision to the Q-List to include classification analysis revisions and to incorporate the Management Control List as an appendix to the Q-List. As a result of this revision, a document action request has been issued to delete the Management

Control List as a separate document. Another major revision to the Q-List is also in progress that will align the Q-List with the current System Description Document organization of structures, systems, and components, which is based on a functional classification analysis of the proposed repository structures, systems, and components. This analysis is expected to be approved next reporting period.

Forecast: Design basis event analyses will continue to be defined and coordinated to evaluate the response of repository structures, systems, and components to potential design basis events to support the designs for the MGDS and Waste Package and to support the establishment of QA classifications for repository items important to radiological safety or waste isolation. A full scope revision of the Q-List will be issued to reflect the current repository design, as defined in the MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b) as modified through the Controlled Design Assumptions Document (CRWMS M&O, 1996c). In addition, formal QAP-2-3 Classification of Permanent Items classification analyses are expected to be performed for Bin 3 system description documents. The results of the analyses from the Design Basis Event Integrated Task Team effort (see Section 4.2.1 of this progress report) will form the basis for QA classifications of repository structures, systems, and components. These results will be included in future Q-List revisions.

2.4 PROGRAM OUTREACH

The objective of Program Outreach is to ensure open and informative interactions with the public and Project stakeholders in accordance with the Nuclear Waste Policy Act and the Secretary of Energy's Public Participation Policy. To comply with these requirements, the Project conducts interactions with the State of Nevada, the public and public interest groups, the Nevada business community, affected counties, local government agencies, the legislature, and the media. The approach to these interactions can be broken into three areas of emphasis: stakeholder involvement, public outreach, and product development. The following are means by which the Project interacted with, and provided information to, the public and Project stakeholders.

Project staff completed a milestone deliverable, the "1996 Year in Review" video, which highlights activities accomplished on the Project during the last fiscal year. The staff also completed and submitted the FY 1997 Yucca Mountain Project Stakeholder Plan Update. The Plan describes the opportunities planned for involving the public in Project activities through use of meetings, workshops, and other forums associated with the Project.

During this reporting period, 89 tours of Yucca Mountain, including two public open house tours, were conducted for approximately 1249 members of the public and other interested parties. Tours of special interest included the Public Broadcasting System, British Broadcasting, the Associated Press, Waste Management '97, the Nuclear Energy Institute, members of Congress, and Congressional staff members.

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Numerous programmatic and technical workshops, presentations, and meetings were held in various Nevada communities to provide current information to the public regarding site characterization progress. The staff held individual meetings with officials of the affected units of government to discuss Project issues and provide updates of Project activities. The staff also held a teleconference meeting with the affected units of government and other interested members of the public on February 20, 1997, to discuss the proposed FY 1998 budget for OCRWM. A total audience of approximately 8300 people attended 110 speaking presentations during this period, which included 13 technical presentations, 30 general Project overviews, and 67 educational presentations.

Approximately 5800 individuals visited the Yucca Mountain Science Centers, located in Las Vegas, Beatty, and Pahrump, Nevada, during this period. At these centers, Project staff presented two Yucca Mountain Speaker Series lectures; 13 teacher workshops; 3 YMCA workshops; and 7 environmental, energy, and geology workshops for fifth-graders. Other events under the Educational Program included two geology walk-n-talks, two Cadette Girl Scouts workshops, one Brownie Troop workshop, one Cub Scout workshop, one Girl Scout Geology Merit Badge Workshop, and one Science Bowl competition. Project staff also participated in Net Day '96, and set up and staffed exhibits for Scout Expo '96 and the Futures Expo on career opportunities. Project personnel participated in ceremonies opening the Net Day '96 initiative in Nevada. The initiative was to connect schools to the Internet and had been supported by Project personnel through contributions and volunteer labor. The Project also revised the Geology Overview. For the Beatty Science Center, staff composed an album of historic photos of early native Americans for use in new cultural resource activities.

The Yucca Mountain Site Characterization Project Home Page on the Internet was accessed over 300,000 times this period by various national and international business, educational and government entities, and members of the public.

The Project researched, created, revised, modified, and distributed a variety of public information products, responded to external information requests, and operated the Civilian Radioactive Waste Management Information Center toll-free number.

<u>Forecast</u>: Project interactions with public and Project stakeholders will continue as needed.

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CHAPTER 3 - SITE PROGRAMS

INTRODUCTION

The Site Characterization Prog. am (Program) consists of 13 scientific investigation programs comprising more than 100 studies. Through the studies, information about the natural features and processes of the Yucca Mountain site is collected, analyzed, and synthesized. This information is needed to determine whether the Yucca Mountain site is suitable for a repository and, if so, to support repository design and application for the license that would authorize construction of a repository. The information collected will be used to determine (a) whether a repository can be constructed and operated at the site while ensuring the health and safety of the ...c, and (b) whether the waste emplaced in the repository will remain isolated from the ...here for thousands of years.

Background

The original testing and analysis approach, as envisioned in the 1988 Site Characterization Plan (SCP) (DOE, 1988), was intended to identify and evaluate significant risks and uncertainties in scientific hypotheses before the license application was prepared and submitted. That testing and analysis program was based on the information requirements in the disposal regulations (10 CFR Part 60) and in the siting guidelines (10 CFR Part 960). The purpose of the SCP was to ensure that sufficient information would be collected to determine the suitability of the site, design a repository, assess the performance of the repository system, and develop an adequate license application for construction authorization. The SCP approach included the flexibility to modify the program as new data confirmed or refuted the importance of specific items of information (see, for example, p. 8-0-1 of DOE, 1988).

Progress in surface and subsurface investigations, especially over the last four and a half years, has allowed the scope of site characterization to be reduced. Observations of the natural system and site data collected since 1978 have allowed many site conditions to be characterized, conceptual models to be advanced, and uncertainties to be bounded. Along with assessments of expected repository performance and continued development of a strategy to contain and isolate the waste, the U.S. Department of Energy (DOE), has used the results from site characterization activities to develop a long-range plan (CRWMS M&O, 1996a) that narrows the scope of the testing and analysis program to focus on the relatively few remaining scientific and technical uncertainties that are important to the design and long-term performance of the potential repository. Recognizing the changes and the need for traceability in the site characterization program, the DOE initiated an effort to document the changes to the program since the SCP was issued. Summary information at the investigation and study level (i.e., 8.3.1.x.x) is included in Appendix A of this progress report.

Data from observation, testing, design and modeling activities, plus published information from non-Yucca Mountain Site Characterization Project (Project) sources will provide the technical basis for the viability assessment, the suitability determination, the Secretarial recommendation, and the license application. By the end of fiscal year (FY) 1998, Project staff expects to provide sufficient information regarding the technical questions about the characteristics of the site and the expected performance of the repository system to support the viability assessment.

Interrelationships with Other Program Elements

The site investigations program is designed to provide information on the geologic, hydrologic, geomechanical, geochemical, climatological, and meteorological characteristics of the site. Information is gathered using a variety of field and laboratory studies The information collected under the site investigations program will provide the basis for the site viability assessment and also will be used by other Program elements to support the Site Recommendation Report and license application.

Systems engineering determines various requirements to ensure that (a) site characterization activities do not compromise the design and construction of the Mined Geologic Disposal System and (b) that construction activities do not compromise site characterization activities. Information on the three-dimensional structural and stratigraphic characteristics of the potential repository block is needed for repository design. In particular, faults will affect repository layout, and tectonics studies are needed to establish seismic design parameters. Likewise, site characterization provides information needed to construct the Exploratory Studies Facility (ESF) and establish controls needed to prevent potential interference between tests. Design organizations identify information needs related to ESF and repository design, and establish controls to prevent potential interference between construction and tests. Particularly important items of information for design and construction are geotechnical properties of the rock and seismic hazard assessments.

Site characterization activities provide information vital to waste package design and material selection, including information on climatology; infiltration; and the unsaturated zone, including bounds on infiltration and percolation flux rates and water chemistry. The site investigations program also provides data on the effects of heat on the physical behavior of rock, on the chemical composition of the water, on matrix and fracture flow in the potential host rock for consideration in waste package design, and on relatively humidity. Design issues pertaining to percolation flux and relative humidity are discussed in Chapter 5 of this progress report. Performance assessment issues pertaining to percolation flux and relative humidity are discussed in Chapter 6 of this progress report.

Significant Results During this Reporting Period

The following numbered items describe significant results from various studies during the reporting period. Generally, the results are listed in three subject areas: hydrologic and pneumatic and transport properties, geology and rock properties, and climate.

Hydrologic, Pneumatic and Transport Properties

1. In the ESF, chlorine-36 from weapons testing occurs in a few distinct fractured and/or faulted zones, indicating that at least a detectable amount of the water at depth is less than 50 years old. Initial analysis suggested that chlorine-36 from weapons testing

appeared to be associated with major faults mapped at the surface. No samples containing chlorine-36 that is unambiguously from weapons testing have been detected in samples collected more than 4500 m from the north portal.

During this reporting period analysis (Levy et al., in prep.) of the distribution in the ESF of chlorine-36 from weapons testing appears to depend on (a) the presence of faults that cut the Paintbrush nonwelded (PTn) hydrogeologic unit; (b) the magnitude of surface infiltration; and (c) the structural features that produce downward lateral spreading away from fault zones (Section 3.1.5, Activity 8.3.1.2.2.2.1).

In the Northern Ghost Dance Fault Alcove, higher porosities occur on the western side (hanging wall) of the Ghost Dance fault, and lower porosities occur on the eastern side (foot wall) of the fault. Sample results indicate that in situ saturation and saturated conductivity may be influenced by the zeolite content of the rock (Section 3.1.7, Activity 8.3.1.2.2.3.1).

2. Recent temperature data collected from two boreholes in Pagany Wash may indicate the passage of an infiltration front through the Tiva Canyon Tuff and upper half of the PTn at UE-25 UZ#5.

   Instabilities and unusual oscillations in temperature and water potential data in USW NRG-7a may be the result of air circulation within the Tiva Canyon Tuff that could be driven by density differences related to topography.

   Changes in temperatures and pneumatic pressures from all Topopah Spring Tuff instrument stations in boreholes USW SD-12, USW NRG-7a, and USW SD-7 are apparently responses to ESF construction operations. In USW SD-12, data from the deepest instrument station (situated below the perched-water zone) indicate an atmospheric loading or strain-induced response associated with the synoptic signal component present in the pressure record. This finding indicates that the perched-water zone is of limited extent. Temperature changes attributable to ESF construction have not been observed in other monitored boreholes at Yucca Mountain (Section 3.17, Activity 8.3.1.2.2.3.2).

3. Rock temperatures near the tunnel boring machine were observed to change spatially and temporally, and the changes have been related to evaporation from the rock surfaces. Recent estimates indicate that ESF ventilation operations effectively remove the equivalent of approximately 200 mm/yr (±100 mm/yr) of moisture. Ventilation effects preclude measurements or observations of natural seepage along the ESF, because modern day percolation flux that would be observed as seepage is estimated to be significantly less than the ESF dryout rate.

   Initial values of water potential measured in the south ramp of the ESF were considerably greater than in situ measurements in surface-based boreholes in similar zones of the PTn. The greater water potential values suggest a higher percolation flux through the PTn than would have been estimated using the surface-based borehole data. However, sensors installed near the ESF wall are drying out at a rate equivalent to 0.5 to 1 mm/day (180 to 365 mm/yr). This rate is supported by the dryout estimated from changes in vapor density calculated using temperature, relative humidity, and wind-speed data. Matrix flux in the PTn using this new data has not yet been estimated (Section 3.1.9, Activity 8.3.1.2.2.4.2).

4. Pneumatic monitoring of boreholes in the Upper Tiva Canyon Alcove showed no differential pressure among three boreholes, and between the boreholes and atmospheric pressure These observations confirm the large fracture permeability of the Tiva Canyon Tuff. In addition, gaseous-phase carbon-14 data indicate a large degree of mixing between atmospheric and rock gases. Comparisons of matrix air permeability values for the Tiva Canyon Tuff indicate that the tuff is isotropic at shallow depths. However, comparison of pneumatic permeability values in the Tiva Canyon Tuff lower lithophysal zone in the ESF with surface values indicates that the zone in the ESF is anisotropic and has a horizontal-to-vertical permeability ratio of approximately 10:1. Tracer tests in the Tiva Canyon Tuff indicated adsorption of the gaseous tracer in the fault breccia and tortuosity in the tuff (Section 3.1.8, Activity 8.3.1.2.2.4.4).

5. Data from tritium analyses of pore water extracted from rock samples collected from various locations in the ESF have been used to determine the spatial distribution of percolation flux. Tritium detected in the vicinity of the Upper Tiva Canyon Alcove is consistent with the concept that water percolates readily from the land surface to the top of the PTn. Alternative interpretations of the data do not preclude the possibility that young water may have percolated below the PTn into the Topopah Spring Tuff middle nonlithophysal zone at three additional locations in the ESF (Section 3.1.8, Activity 8.3.1.2.2.4.8).

6. Pneumatic monitoring in the Bow Ridge Fault Alcove indicated high permeability in the Tiva Canyon Tuff, Bow Ridge fault breccia, and pre-Rainier Mesa Tuff bedded tuff. Tritium values in matrix water indicate that water less than 50 years old has percolated downward along the Bow Ridge fault and mixed with older water in the rock matrix.

   Air-injection testing in a borehole drilled from the Northern Ghost Dance Fault Alcove showed that the permeability of the Ghost Dance fault zone is more than ten times greater than that of the surrounding tuff. Also, because water was redistributed during the single-hole and cross-hole air-injection tests, the capillary pressure of water held in fractures must be less than one atmosphere (Section 3.1.8, Activity 8.3.1.2.2.4.10).

7. Pore water from the unsaturated zone in the PTn has significantly greater concentrations of major ions and dissolved solids than does perched water or saturated zone water. Recharge of perched or saturated-zone waters, therefore, apparently requires rapid flow through fractures or more permeable regions in the unit to avoid mixing with the chemically concentrated water contained within the PTn. These

interpretations are consistent with tritium and chlorine-36 contents of samples from the deep unsaturated zone at Yucca Mountain.

Deuterium and carbon-14 data from pore water in the Topopah Spring Tuff have been interpreted as indicating a post-glacial age (2,000 to 10,000 years old) for the pore water. Corrected carbon-14 data indicate perched water in boreholes USW NRG-7a, USW UZ-14, and USW SD-9 is from 2,150 to 6,260 years old (Section 3.1.11, Activity 8.3.1.2.2.7.2).

8. Using temperature differences between the saturated and unsaturated zone, estimates of unsaturated zone percolation in three boreholes were, respectively, 9.1 mm/yr at USW H-3, 23.5 mm/yr at USW G-1, and 4.9 mm/yr at USW G-3. However, using temperature data from the unsaturated zone alone (Topopah Spring Tuff and the Calico Hills Formation), estimates of percolation flux were 1.8 mm/yr at USW G-1 and 4.6 mm/yr at USW G-4. The difference in the two percolation flux estimates for USW G-1 may be reflecting nonvertical flow in the lower part of the unsaturated zone. Alternatively, the difference could indicate that, even without obvious evidence in the temperature profile, nonconductive transport of heat in the saturated zone is introducing bias into estimates of the unsaturated-zone percolation flux. If the latter is true, more reliable estimates of the percolation flux might be obtained if heat flows are determined using only data from the unsaturated zone.

   A Project initiative titled "Unsaturated-Zone Model Expert Elicitation" began during the reporting period. In general, experts estimate an average percolation flux of 5 to10 mm/yr within the potential repository area based on results of infiltration studies, borehole temperature data, and geochemical isotopic data (Section 3.1.13, Activity 8.3.1.2.2.9.1).

9. Borehole monitoring of water levels in the saturated zone has indicated no seasonal water-level trends in any of the monitored intervals. Regional ground-water withdrawals did not appear to cause water-level changes. Most annual water-level fluctuations were attributed to responses to barometric-pressure changes and earth tides (Section 3.1.14, Activity 8.3.1.2.3.1.2).

10. A long-term pumping test at the C-hole complex has estab[illegible] large-scale structural control on flow in the saturated zone southeast of the poten[illegible]ry site. The test also confirmed hydraulic connection between the C-holes an[illegible] 25 ONC#1 and USW H-4. The overall cone of depression for the test is elongated along the axis aligned in a west-north-west direction where recent geologic mapping indicates discontinuous faults with associated fractures. The faults and fractures probably are the reason for the hydraulic connections between the C-holes and both UE-25 ONC#1 and USW H-4.

11. Recently completed research about chelated transport of iron has shown that chelation significantly affects the transport of iron in crushed tuff. Therefore, there is strong potential for enhanced transport of radioactive wastes by chelation.

12. Calcite and opal veins from the Tiva Canyon Tuff, the Paintbrush nonwelded hydrogeologic unit, and the upper part of the Topopah Spring Tuff have low uranium-234/uranium-238 ratios similar to surficial calcite. The data are interpreted as evidence that large volumes of percolating water would strip available uranium-234 resulting in low ratios. However, samples from the repository horizon have much higher ratios. The higher ratios are interpreted to indicate that relatively small volumes of percolating water reach the repository horizon.

    Reactive transport simulations were conducted for flow regimes that may be expected near the boiling front under conditions in which saturation has been achieved. Preliminary results indicate that the primary concern for hydrologic properties is the formation of cristobalite and/or calcite plugs in fractures in which flux exceeds about 100 mm/yr. The results indicate that under these conditions, precipitation of solids at the boiling front could seal fractures in less than 100 years. Where the movement of the front is slow, fracture sealing could restrict fluid movement at the boiling front. The results also suggest that drainage of fluid through pillars between emplacement drifts may constrain flow because of precipitation from cooling water as the flow enters regions of lower temperatures (Section 3.2.2, Activity 8.3.1.3.2.1.3; Section 3.14, Activity 8.3.1.20.1.1.4).

<u>Geology and Rock Properties</u>

1. Recent drilling in the crowns of the Northern Ghost Dance Fault Alcove and the Southern Ghost Dance Fault Alcove has confirmed that the contact between the Topopah Spring Tuff upper lithophysal zone (Tptpul) and the middle nonlithophysal zone (Tptpmn—repository horizon - see Buesch et al., 1996a) is within 1 m of the location predicted by the three-dimensional lithostratigraphic model. Video logs and core from a horizontal hole, drilled from the Northern Ghost Dance Fault Alcove access drift showed that the main trace of the Ghost Dance fault is located about 154 m east of the ESF main drift. The geometry of distributed fractures in the hanging wall (west side) and the foot wall (east side) is consistent with observations on the surface. Projection of the surface trace of the fault to the location of the main trace at the ESF level indicates that the fault is nearly vertical (Section 3.3.3, Activity 8.3.1.4.2.1.1).

2. Mapping of the central block of Yucca Mountain (Day et al., in press) has shown that numerous intrablock faults that are a narrow zone at depth, such as the Ghost Dance, Abandoned Wash, and Busted Butte faults, "horsetail" toward the surface into a series of bifurcating faults. Mapping during the reporting period has shown that this pattern is maintained for block-bounding faults. The results of this work can be applied directly to understand the foot wall deformation associated with the Solitario Canyon fault, which bounds the western margin of the repository area. Bulk permeability is generally proportional to deformation; so if a reliable method to project changes in deformation can be developed, it may be possible to identify areas of potentially increased permeability that would increase the amount of percolation flux (Section 3.3.4, Activity 8.3.1.4.2.2.1)

3. Geotechnical characterization of the main drift focused primarily on rock-mass quality and mechanical properties. For the main drift, the averages of all rating systems yield rock quality ratings of poor to fair. Between ESF Stations 28 + 00 (2800 m) and 37 + 00 (3700 m) rock quality was fair and decreases to the south (Section 3.3.4, Activity 8.3.1.4.2.2.4).

4. Preliminar . esults from the single heater test indicate that (a) convective heating effects are smaller than predicted from the thermohydrologic-hydrologic analyses, possibly because of vapor escaping the block or from heat loss through fracture systems not accounted for in the modeling; (b) at temperatures above boiling, temperature predictions agree quite well with measured values; (c) the rock mass thermal expansion coefficient is 10 to 20 percent less than the thermal expansion coefficient of the intact rock; (d) the rock mass modulus does not yet show any significant effect from temperature; and (e) the rock bolts installed in the heated zone of the test show greater reduction in tension load than bolts installed in rock remaining at ambient conditions. The tension load is reduced because the expansion of steel in the rock bolts is greater than the expansion of the rock in the temperature range of the test (Section 3 11.5, Activity 8.3.1.15.1.6.2).

5. Data from geotechnical instruments monitoring ground-support conditions and drift convergence have identified no significant concerns regarding the integrity of the ground support or the stability of the opening.

6. A report describing the results of foam-rubber modeling of normal-fault earthquakes provides evidence that near-field ground motions from normal faulting earthquakes may be less than from similar size thrust and strike-slip earthquakes (Section 3.13.9, Activity 8.3.1.17.4.1.2).

7. Observations from the distribution of precariously balanced rocks in the Yucca Mountain region indicate that earthquakes similar in size to the Little Skull Mountain earthquake ($M_L = 5.6$) have not occurred during the last few thousand years.

8. Results from teleseismic tomography studies indicate that no large low-velocity zone exists under Crater Flat or Yucca Mountain. Therefore, no major source of magma is indicated beneath Crater Flat or Yucca Mountain.

Climate

Various paleoclimate and paleohydrologic data were interpreted and synthesized in efforts to establish the nature, duration, amplitude, and magnitude of past climatic changes in the Yucca Mountain region. These interpretations are being used to constrain and bound future climate scenarios for Yucca Mountain, and to guide simulations of possible future conditions with the nested global-regional climate model.

1. The results from diatom and ostracode high-resolution records indicate that limno-climate changes at Owens Lake were extremely rapid; the lake often varied from an

overflowing, freshwater system to a closed, saline or even dry system, in less than a millennium. These types of changes are more typical of interglacial and transitional periods than they are of glacial periods.

Ostracod records for the period from about 55 ka through the Holocene (10 ka to present) show very rapid (decadal) shifts from dry climates, like those of today, to brief periods of warm, wet climates, probably supported by summer rains. The appearance of the full glacial climate (25 ka) was also rapid, but the condition persisted for several millennia.

Radiocarbon-age control has shown that deposition in paleowetlands in the Las Vegas and Indian Springs Valleys was more episodic that originally thought. Packrat (*Neotoma sp.*) midden data imply that the local water table elevation rose above the surface of the valley bottoms only during century- to millennium-long wet periods.

Data from the Owens Lake and from the paleowetlands records indicate that the penultimate glacial period (170 to 130 ka) was substantially wetter than the last glacial period (40 to 10 ka) (Section 3.4.2, Activity 8.3.1.5.1.2).

2. A significant amount of correspondence has been observed between the isotope record at Devils Hole, Nevada, and the paleoclimate records from Owens Lake, California. In addition, both of these records correspond to the marine oxygen isotope records that document global changes in the earth's temperature and ice volume. The correspondence demonstrates that the timing and rate of past climate change at Yucca Mountain coincided with the large, cyclic changes in global climate throughout the Quaternary. Global climates are closely linked to astronomically derived changes in solar insolation and future changes in solar insolation can be determined accurately. Because some understanding has been gained of the magnitude of regional climate change within particular segments of the solar insolation cycle, it may be possible, in general terms, to forecast the magnitude of future climate change in the Yucca Mountain region. In particular, the present-day segment of the insolation cycle resembles the climate that occurred about 400,000 years ago. Therefore, the characteristics of climate that happened about 400,000 years ago may be expected to generally recur in southern Nevada (Section 3.4.5, Activity 8.3.1.5.1.5).

Overview

Table 3-1 lists the 13 scientific programs, and one study that was added after the SCP was issued, that now make up the site characterization effort at Yucca Mountain, Nevada. This table also lists the progress report section number for each program, briefly describes the scope of each program, and references the corresponding section of the SCP.

These programs are discussed in detail in the progress report sections listed in Table 3-1. The discussions in these sections reflect the impact of consolidating study plan work scopes on the individual studies and activities of the SCP. The discussions contain many references to study plans, identify SCP sections for which study plans have not been written, and explain why

Table 3-1. Descriptions and Site Characterization Plan Sections (in parentheses) for Site Programs Described in Chapter 3

3.1	Geohydrology	Investigates surface and subsurface hydrology on both site and regional scales, with ground-water flow system characterization and modeling for both the unsaturated zone and saturated zone (SCP Section 8.3.1.2)
3.2	Geochemistry	Investigates and models rock chemistry and mineralogy, ground-water chemistry, and geochemical behavior of materials along potential radionuclide transport pathways (SCP Section 8.3.1.3)
3.3	Rock Characteristics	Characterizes and models rock stratigraphic and structural features and distributions within the site area, and integrates geophysical and drilling activities to obtain subsurface stratigraphic and structural data (SCP Section 8.3.1.4)
3.4	Climate	Analyzes paleoclimate, paleohydrology, and paleoenvironment, and characterizes modern climate, future climate, and future hydrology (SCP Section 8.3.1.5)
3.5	Erosion	Characterizes modern and past erosion and evaluates the potential effects of future climate and tectonics on erosion (SCP Section 8.3.1.6)
3.6	Postclosure Tectonics	Characterizes tectonic features, such as igneous activity and fault and fold deformation in the Yucca Mountain vicinity, with emphasis on volcanic activity, and analyzes the potential effects of tectonic processes on a potential repository and the site ground water system (SCP Section 8.3.1.8)
3.7	Human Interference	Evaluates the known and potential natural resources in the site area, and the potential for future human intrusion into the site area in search of such resources (SCP Section 8.3.1.9)
3.8	Meteorology	Characterizes the site and regional meteorological conditions of the Yucca Mountain vicinity (SCP Section 8.3.1.12)
3.9	Offsite Installations and Operations	Determines the presence, and potential impacts on the site area, of offsite industrial, transportation, and military installations and operations in the Yucca Mountain vicinity (SCP Section 8.3.1.13)
3.10	Surface Characteristics	Characterizes the topographic characteristics and properties of soil and rock in the site area (SCP Section 8.3.1.14)
3.11	Thermal and Mechanical Rock Characteristics	Determines rock thermal and mechanical properties from laboratory and in situ investigations and characterizes thermal and mechanical stress conditions at the site (SCP Section 8.3.1.15)

Table 3-1. Descriptions and Site Characterization Plan Sections (in parentheses) for Site Programs Described in Chapter 3 (continued)

3.12 Preclosure Hydrology	Characterizes the potential for flooding, determines the location of an adequate water supply for repository construction and operation, and characterizes preclosure hydrologic conditions in the unsaturated zone at Yucca Mountain (SCP Section 8.3.1.16)
3.13 Preclosure Tectonics	Characterizes faults, seismicity and tectonic stress field, and evaluates the potential for faulting, ground motion, and volcanic activity in the site vicinity during the preclosure period (SCP Section 8.3.1.17)
3.14 Altered Zone Characterization	A new activity that was not addressed in the SCP has been created to develop and validate techniques to analyze the performance of the natural system under potential changes resulting from waste emplacement (SCP Section - N/A [new study added after SCP issued])

specific study plans may not be developed. Explanations may include work scope that has been transferred or aggregated into other study plans; work that has been completed and reported; or work for which no study is necessary because data are available from other sources. The status of the study plans is summarized in Appendix G.

Four figures and one additional table have been included to help the reader understand the information presented in this chapter. These figures and the table are referenced in many of the study descriptions.

- Figure 3-1 shows surface features in the Yucca Mountain region.
- Figure 3-2 shows the locations of the proposed repository block and of boreholes in the potential repository area.
- Figure 3-3 shows locations of selected geologic and potential repository features related to significant results reported this period.
- Figure 3-4 shows the ESF and test alcoves and presents a summary of the remaining test activities for each alcove.
- Table 3-2 shows the stratigraphy of Yucca Mountain as determined for various usage classifications.

See the references section (Appendix L) for information about compilation of figures required by Federal Executive Order 12906 (59 FR 71, p. 17671-17674).

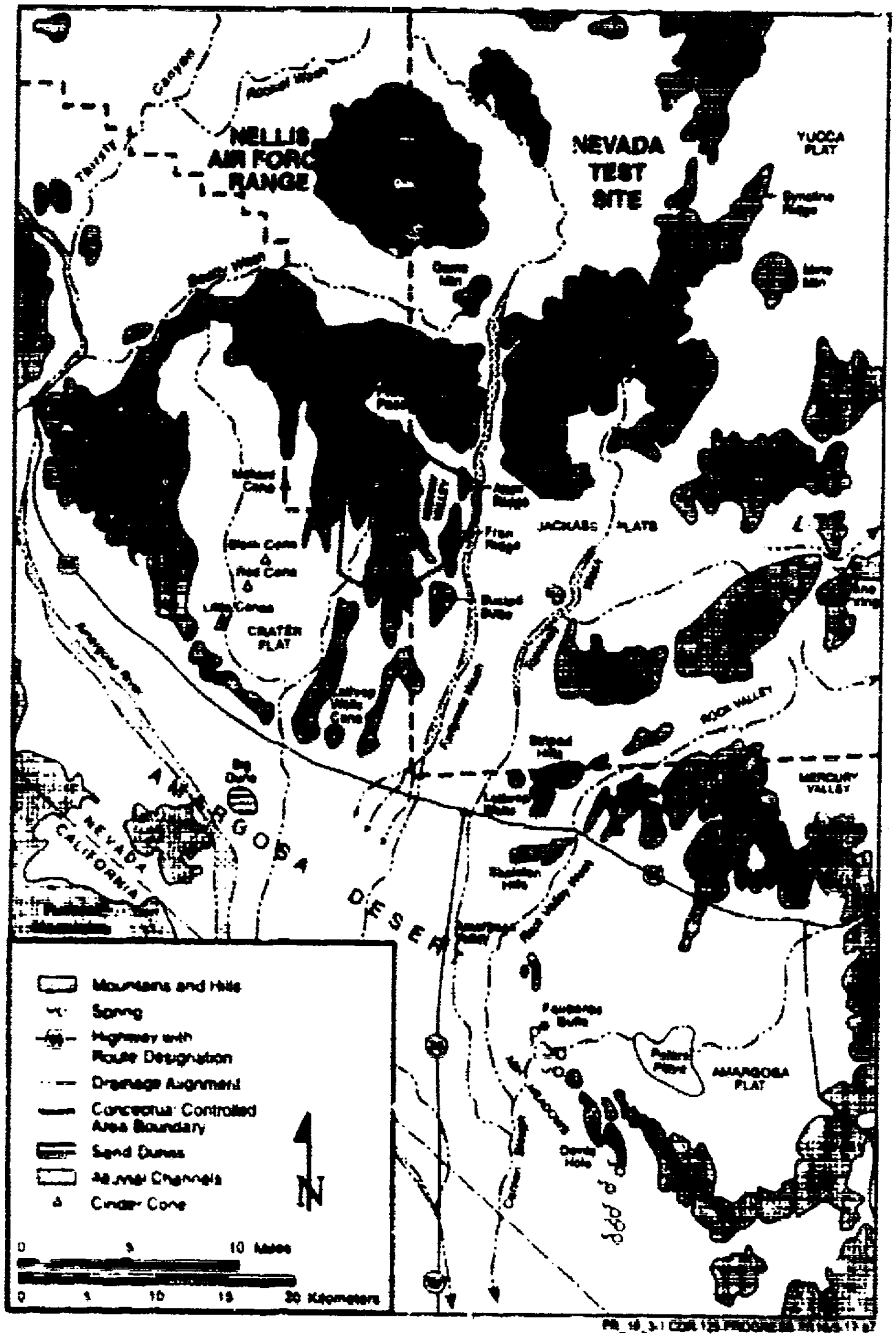
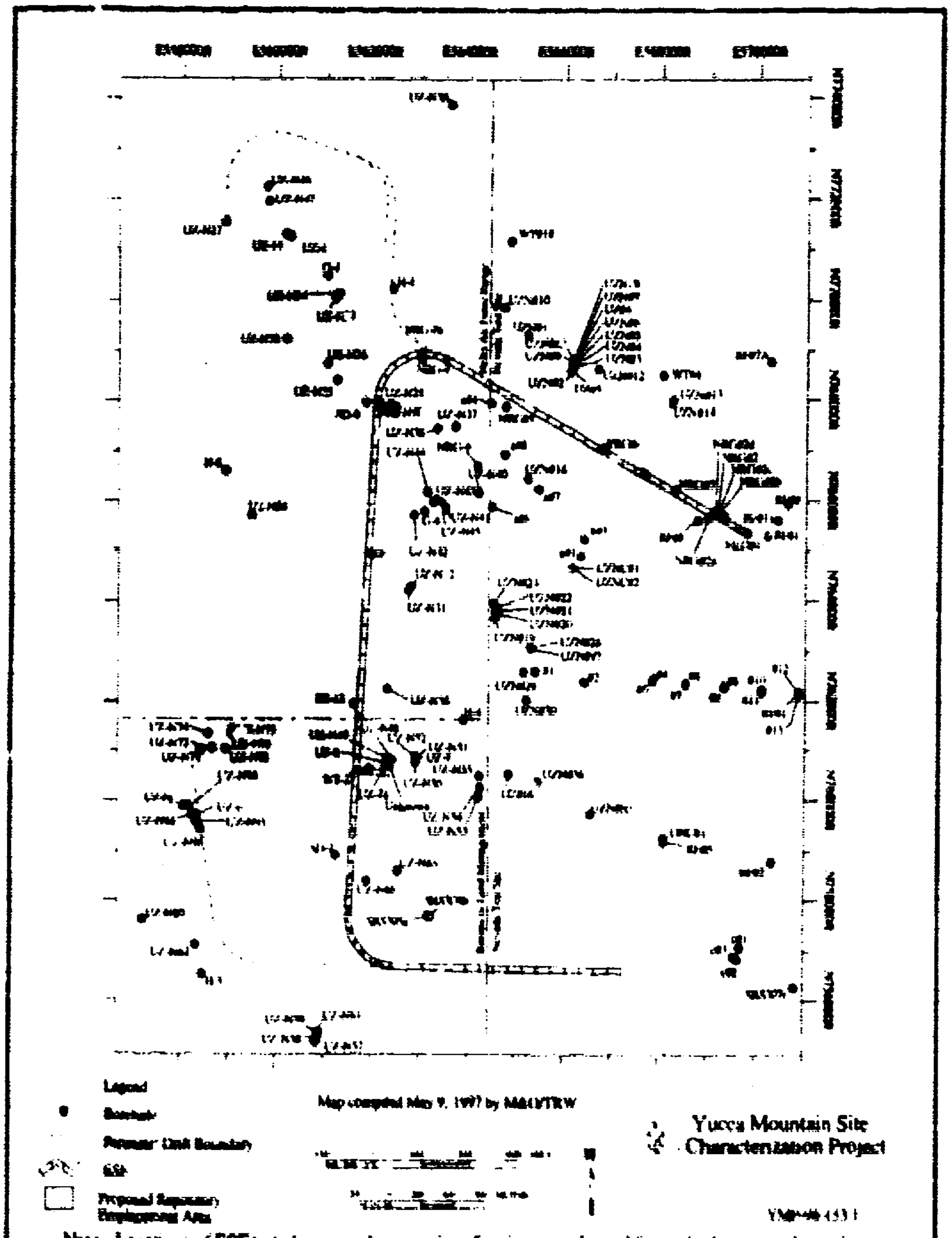


Figure 3-1. Regional Location and Surface Features Map



Note: Locations of ESF test alcoves and summaries of major tests planned for each alcove are shown in Figure 3-4. For location of TBM at the end of the reporting period, see Chapter 7, section 7.3.

Figure 3-2 Map Showing the Exploratory Studies Facility, Potential Repository Block, and Repository Area Boreholes

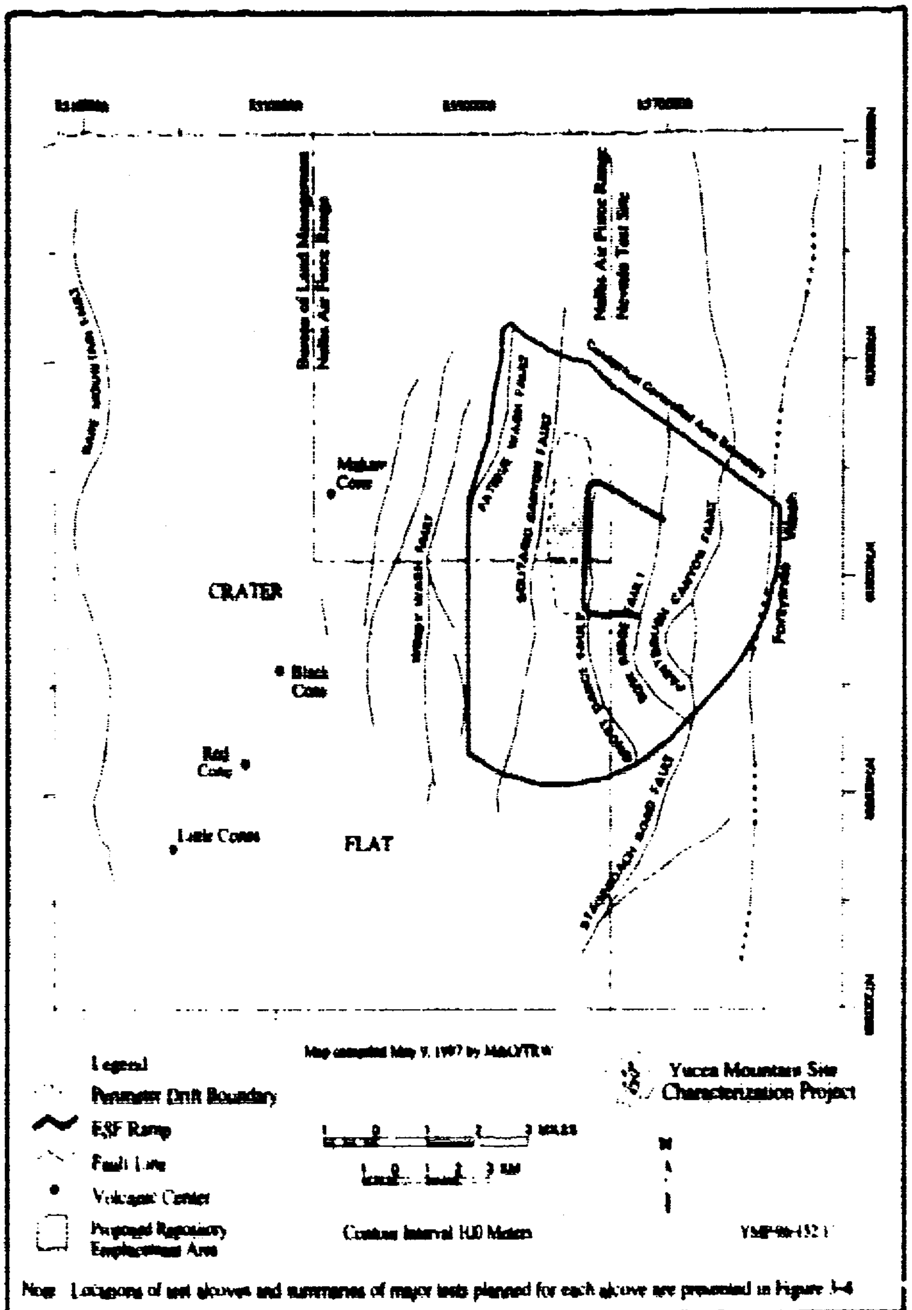


Figure 3-3. Map Showing Locations of Selected Site Features

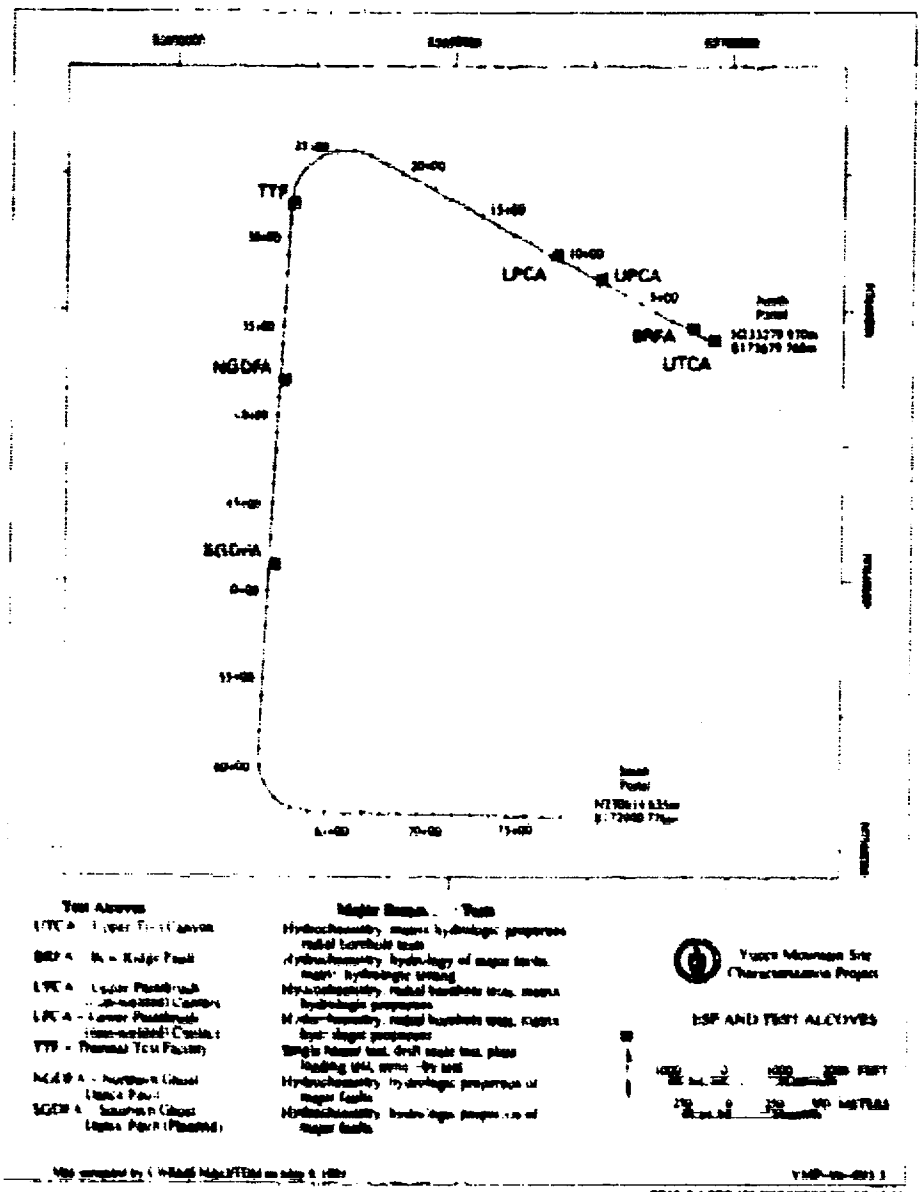


Figure 3-4 Map of Exploratory Studies Facility (ESF) Showing Names and Locations of Test Alcoves and Major Tests Planned for Each Alcove

Table 3-2. 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&C, 1996f)

Group	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
Paintbrush Group	pre-Tiva Canyon bedded tuff	Paintbrush Nonwelded Unit PTn	PTn
Paintbrush Group	Yucca Mountain Tuff		
Paintbrush Group	pre-Yucca Mountain bedded tuff		
Paintbrush Group	Pah Canyon Tuff		
Paintbrush Group	pre-Pah Canyon bedded tuff		
Paintbrush Group	Topopah Spring Tuff	Topopah Spring Welded Unit TSw	TSw1
			TSw2
			TSw3
		Calico Hills Nonwelded Unit CHn	CHn1v
	pre-Topopah Spring bedded tuff		
	Calico Hills Formation		CHn1z
			CHn2z
Crater Flat Group	Prow Pass Tuff	Crater Flat Unit CFu	CHn3z
			PPw
			CFun
Crater Flat Group	Bullfrog Tuff		BFw
			CFMn
Crater Flat Group	Tram Tuff		TRw

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3.1 GEOHYDROLOGY (SCP SECTION 8.3.1.2)

The changes to the Geohydrology Program since the SCP was issued are summarized in Appendix A in Section A.1.1.

3.1.1 Study 8.3.1.2.1.1 - Characterization of the Meteorology for Regional Hydrology

The objectives of this study are (a) to characterize the area surrounding Yucca Mountain in terms of precipitation and its relationship to surface runoff, with particular emphasis on the Fortymile Wash drainage basin, and (b) to provide input into the rainfall-runoff model development effort.

In 1990, the study was modified to emphasize meteorological information needed to develop an infiltration model for Yucca Mountain. At that time, precipitation-runoff modeling was de-emphasized (see Appendix A, Section A.1.1.1 of this progress report).

Closing calibrations for all field instruments used on the weather stations were completed as part of preparing FY 1996 data packages. The weather-station data, collected before disassembly of the remaining two weather stations, were compiled into daily files and technically reviewed.

Closing calibrations for all instruments used in the network of 14 tipping-bucket precipitation gauges were completed, and the timing of 0.1 mm precipitation events was determined. The network of 14 tipping-bucket gauges was transferred to the Environmental Field Program for air quality and meteorology (see Section 3.8 of this progress report). As part of the transfer, all equipment was removed from the field, cleaned, calibrated, and reinstalled for use by the Environmental Field Program. The Environmental Field Program now has assumed responsibility for collecting all meteorological data previously collected under this study.

All meteorological data and supporting documentation from the two weather stations, the tipping-bucket network, and the storage-gauge network have been technically reviewed and submitted to the Records Processing Center and Technical Data Base.

Work under this study was discontinued effective March 31, 1997.

Forecast: All future meteorological data collection will be conducted under the Meteorology Program (SCP Section 8.3.1.12) described in Section 3.8 of this progress report.

3.1.2 Study 8.3.1.2.1.2 - Characterization of Runoff and Streamflow

The objectives of this study are to (a) collect basic data on surface-water runoff at, and peripheral to, Yucca Mountain and on its hydrologic flow system, (b) use the streamflow data to describe the runoff characteristics of the area and to assess the response of runoff to precipitation; (c) assess the potential for flood hazards and related fluvial-debris hazards; and (d) provide basic data and interpretations of surface-water runoff to investigations that evaluate the amounts and processes of infiltration and ground-water recharge at Yucca Mountain and surrounding areas.

Activity 8.3.1.2.1.2.1 - Surface-water runoff monitoring. The objectives of this activity are (a) to develop needed basic data on the characteristics, magnitudes, frequencies, and timing of surface-water runoff to develop an understanding of the relation between specific runoff events and the characteristics of the storms and associated precipitation; and (b) to develop a streamflow data base adequate to provide the necessary calibration data for precipitation-runoff modeling efforts for the regional study area, and site-scale water-balance and infiltration modeling.

Monitoring continued at the three continuous-recording streamflow gauges on Fortymile Wash at the Narrows, Fortymile Wash near well UE-25 J#13, and Fortymile Wash near Amargosa Valley. During November 1996 several storms moved across southern Nevada, and minor runoff resulted in the Yucca Mountain area. Small amounts of flow were reported at the Fortymile Wash gauges at the Narrows and near well UE-25 J#13. Stages were too low, however, to be recorded by the monitoring equipment.

Records of data collected during water year 1996 for the three continuous-recording streamflow gauges were processed for publication in the U.S. Geological Survey (USGS) annual water-data report for Nevada

Activity 8.3.1.2.1.2.2 - Transport of debris by severe runoff. The objective of this activity is to document, both quantitatively and qualitatively, the characteristics of debris transported by intense surface runoff.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Monitoring will continue at the three continuous-recording streamflow gauges. Data records for water year 1996 will be published in the annual water-data report for Nevada.

3.1.3 Study 8.3.1.2.1.3 - Characterization of the Regional Ground-Water Flow System

The objectives of this study are to further define the distribution of hydraulic properties of the regional ground-water flow system and to use hydrologic, hydrochemical, and heat-flow data to determine the magnitude and direction of ground-water flow

Activity 8.3.1.2.1.3.1 Assessment of regional hydrologic data needs in the saturated zone. The objective of this activity is to prioritize data needs for use in the regional ground-water flow description.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

<u>Activity 8.3.1.2.1.3.2 - Regional potentiometric-level distribution and hydrogeologic framework studies</u>. The objectives of this activity are to determine the potentiometric distribution within the regional ground-water flow system, and to characterize the hydrogeologic framework of the regional ground-water flow system to support reliable estimates of ground-water flow direction and magnitude within the saturated zone.

No progress was made during the reporting period; this was an unfunded activity.

<u>Activity 8.3.1.2.1.3.3 - Fortymile Wash recharge study</u>. The objective of this study is to determine to what extent (quantitatively, if feasible) Fortymile Wash has been a source of recharge to the saturated zone under present and past conditions.

No progress was made during the reporting period; this was an unfunded activity.

<u>Activity 8.3.1.2.1.3.4 - Evapotranspiration studies</u>. The objective of this activity is to improve estimates of ground-water discharge by evapotranspiration in the Amargosa Desert to provide boundary-condition data for regional ground-water flow models.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: This study was not funded for FY 1997 and no work is planned.

3.1.4 Study 8.3.1.2.1.4 - Regional Hydrologic System Synthesis and Modeling

The objectives of this study are to synthesize the available data into a model and make a qualitative analysis of how the system is functioning, and to represent quantitative observations of hydrogeologic data pertaining to the ground-water flow system in a comprehensive numerical model of ground-water flow.

<u>Activity 8.3.1.2.1.4.1 - Conceptualization of regional hydrologic flow models</u>. The objectives of this activity are to synthesize available data into a conceptual model that incorporates alternative hypotheses and/or existing hypotheses, to make a qualitative analysis of how the regional and subregional ground-water flow systems function, and to describe the regional saturated zone ground-water flow system.

No activity occurred during the reporting period. Work scope for this activity has been transferred to Activity 8.3.1.2.1.4.4 (see Section A.1.1 of Appendix A of this progress report).

<u>Activity 8.3.1.2.1.4.2 - Subregional two-dimensional areal hydrologic modeling</u>. The objective of this activity is to improve estimates of regional ground-water flow by updating an existing two-dimensional, subregional, parameter-estimation model by incorporating additional hydrogeologic data.

No activity occurred during the reporting period. Work scope for this activity has been transferred to Activity 8.3.1.2.1.4.4 (see Section A.1.1 of Appendix A of this progress report).

Activity 8.3.1.2.1.4.3 - Subregional two-dimensional cross-sectional hydrologic modeling. The objective of this activity is to estimate the ground-water flow direction and magnitude along a potential flow path through the repository block to the accessible environment, and extending into the region, to help test the assumption of horizontal flow.

No activity occurred during the reporting period. Work scope for this activity has been transferred to Activity 8.3.1.2.1.4.4 (see Section A.1.1 of Appendix A of this progress report).

Activity 8.3.1.2.1.4.4 - Regional three-dimensional hydrologic modeling. The objective of this activity is to construct a three-dimensional model of the regional ground-water flow system of Yucca Mountain and vicinity.

The three-dimensional hydrogeologic framework model used to construct the regional ground-water flow model was updated to ensure consistency with the calibrated flow model. Hydraulic-conductivity attributes from the calibrated regional flow model were added to the appropriate framework model grid cells. The revised framework model was reviewed and submitted to the Records Processing Center. Model output of simulated fluxes along the boundaries of the site saturated zone flow model were extracted from the regional model and reformatted for use in the site saturated zone flow model. Calibration of the regional flow model was completed and documented during the last reporting period (Progress Report #15, DOE, 1997e).

Regional flow modeling focused on simulating past- and potential future-climate scenarios using the existing calibrated flow model. Precipitation data for two climate scenarios were received from the National Center for Atmospheric Research, which is conducting the numerical simulations of future climate. The first data set represents results of the simulation of maximum continental ice cover during glaciation, which is believed to have occurred about 21,000 years ago and are believed to represent the wettest, coolest climate possible at the Yucca Mountain site that might occur during the proposed repository performance period. The second data set represents results of the simulation of a possible future climate scenario in which carbon dioxide concentration in the earth's atmosphere has doubled because of the "greenhouse effe[illegible] The simulated precipitation data were reprocessed into distributions of recharge corresponding with the grid for the regional ground-water flow model. This reprocessing was accompli[illegible] by kriging precipitation output from the National Center for Atmospheric Research Gl[illegible] limate Model (see Section 3.1.5 of this progress report) and then applying methods describe[illegible] y Hevesi and Flint (in press). Digitized data input arrays of recharge for the MODFLOWP simulation code were constructed using the reprocessed recharge distributions. Maps of past-discharge areas were constructed and qualitative estimates of discharge rates for each area were provided by Project scientists. These discharge estimates will be used to evaluate and bound simulations of past and possible future climate scenarios. Simulation began of the past and possible future climate scenarios with the regional ground-water flow model.

The results of the regional flow model calibration are found in D'Agnese et al., in press[a]. Additional revisions were made to a second report (D'Agnese et al., in prep.) that documents the estimated regional potentiometric surface. A journal article (D'Agnese et al., in press[b]) documenting regional recharge and discharge areas, using regional distributions of vegetation types, was prepared.

Forecast: The simulation of the past and possible future climate scenarios using the existing flow model will be completed. A draft report documenting the results of those simulations will be completed and submitted for review and approval. The report documenting the calibration of the existing regional flow model will be revised and published. The report documenting the regional potentiometric surface will be submitted for approval and prepared for publication.

3.1.5 Study 8.3.1.2.2.1 - Characterization of Unsaturated Zone Infiltration

The objectives of this study are to determine the effective hydraulic conductivity, storage properties, and transport properties as functions of moisture content or potential, and to determine the present and estimate the future spatial distribution of infiltration rates over the repository block.

The primary goal of the reconfigured study is to define the upper-boundary conditions for the site unsaturated zone hydrologic system, in terms of spatial and temporal distribution of moisture flux, for use in site-scale models of unsaturated zone ground water flow and transport. The integration of hydrologic process models for the surface and near surface environments with mapped surficial-material hydrologic properties and mapped present-day net infiltration rates (in terms of both matrix and fracture flow) will provide the basis for a stochastic-deterministic simulation of future upper boundary conditions, including the probability and magnitude of potential fast pathways of net infiltration. Once the infiltration flux at the upper boundary of the unsaturated zone has been defined, the site-scale model will be used to simulate flow across the potential repository horizon for the set of conditions chosen.

Activity 8.3.1.2.2.1.1 - Characterization of hydrologic properties of surficial materials. The objective of this activity is to characterize the infiltration related hydrologic properties and conditions of the surficial soils and rocks covering Yucca Mountain.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.2.2.1.2 - Evaluation of natural infiltration. The objective of this activity is to characterize present day infiltration processes and net infiltration rates in the surficial soils and rocks covering Yucca Mountain.

Net infiltration is the amount of natural surface-infiltrated water less losses to evapotranspirative and other processes such as circulation of air within the porous rock mass. Study results indicate that the properties and processes most important for determining net infiltration are the timing and amount of precipitation, the storage capacity of the soil (which includes soil depth), the seasonality and amount of evapotranspiration, and the hydrologic properties of the underlying bedrock.

Infiltration Distribution

This part of the natural-infiltration activity was completed in FY 1996. See Progress Report #15

Infiltration Processes

Analysis to confirm the numerical model of net infiltration was conducted using 15 years of precipitation data and 10 years of neutron-hole moisture data. With minor exceptions, the results presented in Progress Report #15 were substantiated by the additional analysis. The analysis indicated that the principal weakness in the infiltration model is the lack of a module to simulate the generation and routing of surface runoff. Although the overall output of the infiltration model may not be affected significantly, the lack of runoff routing information makes the net-infiltration values produced by the model more difficult to evaluate. Any additional changes to the net-infiltration model, such as the addition of runoff routing, likely will increase the estimates of net infiltration above those currently being generated.

Analysis of two climate scenarios was completed for the regional ground-water-modeling study (see Section 3.1.4 of this progress report). One scenario involves a past climatic condition about 21,000 years ago when glaciation was at a maximum, and the other involves a possible future climatic condition when atmospheric carbon-dioxide concentration might be doubled by the "greenhouse effect." The past climatic condition is important to repository performance assessments because it could recur within the next 100,000 years. Simulated precipitation data for each of these climatic conditions were obtained from the National Center for Atmospheric Research Global Climatic Model (see Section 3.4.6 of this progress report). For each scenario, the ratio of the changed climate scenario to the base case (current climate) was used to produce a "change map" of precipitation. Each change map was produced by kriging the 64 Global Climatic Model modeled points (50-km spacing) covering the Yucca Mountain regional study area. The "change m " was used in combination with the present-day regional precipitation map to produce past and future precipitation maps. The kriged precipitation maps indicated a 71 percent increase in precipitation for the past-climate scenario and a 17 percent increase in precipitation for the future-climate scenario. The climate scenario precipitation maps were then used as input to the Maxey-Eakin method for calculating recharge, which is described in Hevesi and Flint (in press). Given that on a regional basis the recharge at Yucca Mountain is 1 to 2.5 mm/yr, the past climate scenario yielded 10 to 25 mm/yr (a ten-fold increase in recharge) and the future climate scenario yielded 2.5 to 5 mm/yr (a two-fold increase in recharge). Because these values are really averaged over the regional study area, they should not be interpreted as an estimate of net infiltration over the area of the potential repository. Estimated infiltration rates at the potential repository location likely would be larger (Flint et al., in prep.) because of additional runoff routing information.

Activity 8.3.1.2.2.1.3 - Evaluation of artificial infiltration. The objective of this activity is to characterize water movement in the surficial materials of Yucca Mountain under controlled conditions. Experiments designed to determine the total water flux, flow velocities, and flow paths will be performed on the geohydrologic surficial units under both present-day precipitation rates and simulated higher rates corresponding to wetter climatic conditions.

With the reconfiguration of the study, this activity is now limited to field validation of the infiltration model described in Activity 8.3.1.2.2.1.2 for simulated wetter conditions.

No progress was made during the reporting period; this was an out-year activity.

Forecast: No additional work is planned under this study for the remainder of FY 1997. A proposal will be prepared to incorporate a stream runoff-routing subroutine into the numerical infiltration model during FY 1998.

3.1.6 Study 8.3.1.2.2.2 - Water Movement Test

The objective of this study is to obtain information from isotopic measurements of soil, tuff, and water samples collected to help quantify the amount of percolation into the unsaturated zone at Yucca Mountain.

Activity 8.3.1.2.2.2.1 - Chloride and chlorine-36 measurements of percolation at Yucca Mountain. The objective of this activity is to help quantify the amount of percolation into the unsaturated zone at Yucca Mountain. The data will be used as part of the input to characterize the movement of water through the unsaturated zone at Yucca Mountain.

Analyses of Exploratory Studies Facility Samples

As of March 1997, sampling to support FY 1997 deliverables was completed up to ESF Station 69 + 42 (6942 m from the ESF north portal), and analytical data were available up to ESF Station 67 + 90 (6790 m). Results of the analysis of chlorine-36 data for 187 samples were used to assess ground-water travel times and identify potential fast paths for infiltrating water (Levy et al., in prep). Rock samples were systematically collected every 200 m throughout the tunnel [every 100 m starting with Station 61 + 00 (6100 m)], and feature-based samples were collected from specific geologic features such as faults, fractures, lithophysal cavities, and subunit contacts. Chlorine-36 from weapons testing occurs in a few distinct fractured and/or faulted zones, indicating that at least a detectable amount of the water is less than 50 years old. Initial analysis suggested that chlorine-36 from weapons testing appear to be associated with major faults mapped at the surface. Levels of chlorine-36 unambiguously from weapons testing have not been detected in any of the 54 samples collected beyond Station 45 + 00 (4500 m).

A report (Levy et al., in prep.) examining the sampling results primarily focuses on the detailed characterization of the petrologic and structural settings of sample sites, particularly the so-called "fast paths" that contain a component of chlorine-36 from weapons testing. The methodology involves examining relevant field evidence for (a) correlations between surface and subsurface structural features; (b) vertical connectivity and characteristics of the fracture networks in the Tiva Canyon welded (TCw), the PTn, and the Topopah Spring welded (TSw) hydrogeologic units; and (c) possible changes in fracture intensity near fault zones. These lines of evidence indicate the distribution of chlorine-36 in the ESF from weapons testing depends on (a) the presence of faults that cut the PTn hydrogeologic unit, (b) the magnitude of surface infiltration, and (c) the structural features that produce downward lateral spreading of flow from fault zones.

1. Presence of faults that cut the PTn hydrogeologic unit. The most important structural control seems to be the presence of faults that create breaks in the PTn. Sample locations in the north ramp and main drift of the ESF that show evidence of chlorine-36 from weapons testing are located in the general vicinity of projections of faults mapped at the surface. These include a block-bounding fault (Bow Ridge fault), a probable strike-slip fault (Drill Hole Wash fault), and smaller, intrablock faults (the Sundance fault). The defining characteristic is that the fault must be large enough to break the PTn; fault type, orientation, and amount of offset have no apparent affect on the presence of chlorine-36. This observation is consistent with the results of hydrologic modeling of percolation through the PTn hydrogeologic unit. Model results required a fault through the PTn for at least a small amount of water with the weapons testing signature to arrive at the middle nonlithophysal zone of the Topopah Spring Tuff (main drift of the ESF).

   In some instances, the locations of ESF samples containing chlorine-36 from weapons testing do not appear to correspond to a fault mapped at the surface; yet the presence of chlorine-36 from weapons testing may still be inferred to be the likely result of a fault that cuts the PTn hydrogeologic unit. One example is a sample site located under Diabolus Ridge, which is not obviously associated with a fault mapped at the surface. However, across the central part of Diabolus Ridge, a gently east-dipping reverse fault is mapped at the surface. This fault has about 7 m of offset at the level of the upper portion of the Tiva Canyon Tuff. Projected downward, this fault cuts the base of the PTn hydrologic unit almost directly above the chlorine-36 sample location at ESF Station 26 + 79 (2679 m). Although various other pathways could be envisioned, the simplest explanation for the presence of chlorine-36 from weapons testing is that the fast pathway includes the fault that cuts the PTn hydrogeologic unit above the sample location.

2. Magnitude of surface infiltration. Limited data indicate a correlation between the chlorine-36 signature of a particular fault zone and the surface location of the fault with respect to spatially distributed infiltration. In the main drift of the ESF, faults that have elevated chlorine-36/chloride values indicating a component of chlorine-36 from weapons testing tend to coincide with infiltration highs as defined by Flint et al. (in prep.). For example, both the Sundance fault and the fault on Diabolus Ridge are relatively minor features at the surface, but their surface traces intersect fractured bedrock on topographically high ridge tops that are free of alluvial cover, conditions that may be conducive to elevated infiltration rates. In contrast, faults that do not have elevated chlorine-36/chloride values have surface traces that tend to coincide with infiltration lows. For example, where the surface traces of the southwestern splay of the Drill Hole Wash fault and the southern part of the Ghost Dance fault overlie the ESF, the faults are exposed in topographically low wash bottoms or side slopes with thick alluvial material, conditions associated with low levels of infiltration. These limited data indicate that in the ESF, even faults with relatively minor offset may be

associated with the presence of chlorine-36 from weapons testing, where they intersect zones of sufficiently high infiltration. Conversely, faults of similar, or even greater, offset that intersect zones of low infiltration may not have elevated chlorine-36/chloride values.

3. Structural features that result in downward lateral spreading of flow away from fault zones. A number of the sample localities along the north ramp of the ESF have elevated chlorine-36/chloride values and are tightly grouped near a single fault or an individual joint. In contrast, ESF sample localities near the Sundance fault with elevated chlorine-36/chloride values form a broad zone encompassing a 300-m distance along the ESF. The Sundance fault is the only known structure mapped in this vicinity that could have served as the pathway through the PTn hydrogeologic unit. Therefore, it is hypothesized that the broad zone of elevated chlorine-36/chloride values north of the fault thus represents downward lateral spreading of flow away from the plane of the fault. Such flow implies a connection between the fault and other structures, either small subsidiary faults or interconnected joints, in the rock mass surrounding the fault.

Levy et al. (in prep.) also describes the attributes of fast pathways that contribute to the formation of distinctive secondary mineral assemblages. Preliminary mineralogic and petrologic analysis of 39 samples in the chlorine-36 data base suggests that fast pathways may have some subtly distinctive mineralogic characteristics. Calcite, a common mineral in the ESF, is even more common in samples containing chlorine-36 from weapons testing, but this criterion alone is insufficient to predict the locations of fast paths. Observations indicate that the calcite deposits in fast-path transmissive features are much thinner than the calcite-opal deposits used for thorium-230/uranium geochronologic studies of mineral deposition and inferred infiltration rates. Opal, unlike calcite, appears to be less common in fast pathways than its overall abundance in the ESF would suggest.

Clay deposits that coat fractures or breccia clasts are assumed to result from aqueous transport of fine clay particles within the fracture network and are unlikely to have been derived from the adjacent rock matrix. The mineralogic data set described in Levy et al. (in prep.) does not indicate a consistent connection between fast paths and the presence of clay/mordenite. Fast-path sites associated with the Sundance fault, however, are correlated with the presence of clay/mordenite deposits. In addition to clay, there are also deposits of coarser particulate materials. The most prominent of these is a deposit in a fracture in the Topopah Spring Tuff with graded bedding that clearly indicates particle settling in a water-filled fracture in the past. This sample has no weapons testing chlorine-36 signature. The material in this deposit was probably transported before the PTn nonwelded tuffs were emplaced on top of the Topopah Spring Tuff. Particulate deposits such as this may relate more to former than to present fast pathways in the Topopah Spring Tuff.

Forecast: The results of this study will be applied to the validation of site-scale hydrologic flow and solute transport models being developed under other activities. Project scientists will continue collecting and analyzing samples from the ESF to test or confirm hydrologic-flow hypotheses made using previous results. Collecting and analyzing of samples from boreholes will continue, as appropriate. The contribution of the in situ produced chlorine-36 (e.g., from soil calcite) to the subsurface concentrations of this isotope will be evaluated.

Existing chloride and chlorine-36 data for boreholes and the ESF will be interpreted in terms of infiltration rates and identification of fast-transport paths from the surface to the sampled depths. Chlorine-36 distributions will be correlated with lithology, structural features, and past climates. Corrected ground-water travel time estimates will be produced using data from chlorine-36 samples from boreholes and the ESF. Chlorine-36 analyses will be synthesized. Hydrologic analysis and simulation activities will be performed using the Finite Element Heat and Mass (FEHM) code to help interpret chlorine-36 data. Data and interpretations will be provided to hydrologic and solute transport modelers (a) to ensure appropriate use of isotopic results in model validation exercises and in the various hydrology reports, and (b) to ensure consistency in interpretation of results from this activity with those geochemical and isotopic results obtained from other activities.

Elevated chlorine-36 occurrences, possibly from weapons testing, will be corroborated using other environmental tracers such as iodine-129 and technetium-99. Input functions for these tracers will be developed, as appropriate, to compare their timing and magnitude relative to that for chlorine-36 attributed to weapons testing.

3.1.7 Study 8.3.1.2.2.3 - Characterization of Percolation in the Unsaturated Zone—Surface-Based Study

The objectives of this study are to determine the present in situ hydrologic properties of the unsaturated zone hydrogeologic units and structural features; to determine the present vertical and lateral variation of percolation flux through the hydrogeologic units and structural features; to investigate the relation between present flux and past climatic conditions; and to determine the effective hydraulic conductivity, storage properties, and transport properties as functions of moisture content or potential.

Locations of most of the boreholes identified in the activities included in this study are shown in Figure 3-2. Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpmn and Tptpll) are found in Buesch et al. (1996a).

Activity 8.3.1.2.2.3.1 - Matrix hydrologic properties testing. The objectives of this activity are to conduct laboratory measurements of rock-matrix hydrologic properties on borehole and ESF samples from all hydrogeologic units in the unsaturated zone to characterize the spatial distribution of hydrologic properties within the unsaturated zone.

This activity has been focused to provide two primary information products: (1) a data base of rock matrix hydrologic properties from a series of boreholes that is qualified for site characterization use, and (2) an interpretation of the hydrologic properties of specific hydrogeologic units. Hydrogeologic units have been determined on the basis of similarities within and differences between lithostratigraphic units for the purpose of developing hydrologic flow models.

A study was conducted in the ESF main drift to assess the spatial variability of properties within the Topopah Spring Tuff middle nonlithophysal zone (Tptpmn), the host rock for the potential repository. The study addressed several technical concerns of the unsaturated zone site characterization and performance assessment modelers. Typically, rock properties are input into numerical model layers in one of three ways: (1) as a mean value (either arithmetic, geometric, or power-law mean depending on the property); (2) as a random distribution based on a probability-distribution function determined from all measured values of the property; or (3) as the measured mean value and variance of the rock property as an initial condition for inverse modeling to match a hydrologic property, such as saturation. The properties of the rocks also can be distributed spatially using geostatistical analyses for those properties for which there are spatially distributed estimates, such as porosity. Overall, the Tptpmn has low matrix permeability and high in situ saturations.

Seventy rock samples were collected systematically every 40 m along the wall of the ESF main drift to investigate the possible presence of preferential matrix-flow zones within the Tptpmn. Samples were collected along a transect from the top of the unit at ESF Station 28 + 00 (2800 m) through the base of the unit and into the lower lithophysal zone (Tptpll) at ESF Station 58 + 00 (5800 m). To date, only porosity, bulk density, and particle density have been measured. These data have been compared with the lithostratigraphic descriptions, borehole samples collected from nearby boreholes, mineralogy surveys, chlorine-36 measurements, and line fracture surveys. From top to bottom, porosity along the transect within the middle nonlithophysal zone ranges from 9 to 16 percent in an upper lithophysal subzone, porosity then decreases to about 8 to 12 percent through the primary nonlithophysal subzone, increases to 10 to 13 percent in a lower lithophysal subzone (which at this location does not contain lithophysal cavities but merely a higher matrix porosity and which corresponds to the closely spaced vertical fracture zone informally named the Broken Limb fracture zone). Finally, porosity decreases to 7 to 11 percent as the middle nonlithophysal zone transitions into the lithophysal rocks (Tptpll) below. These changes in porosity can be explained relatively well based on lithostratigraphy and vapor-phase activity during cooling of the tuffs and are supported by measurements of samples from nearby boreholes. The significant interpretation of these data is that there are trends indicating areas of higher and lower porosity that are not randomly distributed where water flow may be concentrated in the matrix. Interpretations of mineral-coatings data that might corroborate the matrix-flow hypothesis presently are inconclusive. There is, however, evidence of a relationship between the color in the rock unit and the matrix properties: gray is associated with higher matrix porosity and pink with more massive rock with lower matrix porosity.

Samples collected from various boreholes in ESF alcoves were processed for matrix hydrologic properties in support of borehole pneumatic monitoring, air-permeability testing, gas sampling, moisture-monitoring, and percolation studies. Borehole samples were collected and processed from the Bow Ridge Fault Alcove, Upper Paintbrush Tuff Contact Alcove, Lower Paintbrush Tuff Contact Alcove, and Northern Ghost Dance Fault Alcove. Results of analyses of samples from the Northern Ghost Dance Fault Alcove indicated higher porosities (9 to 11 percent) on the western side (hanging wall) of the fault and lower porosities (8 to 9 percent) on the eastern side (footwall). Only one sample, collected from the edge of the fault on the hanging wall, exhibited a saturated water content. This sample may be indicative of water flow within the fault, but more likely is evidence of a very localized condition caused by a capillary barrier at the end of the fault zone.

Additional analyses were performed to correlate matrix hydrologic properties of borehole core samples with mineral percentages. Results of analyses indicated that an estimate of the influence of altered minerals on hydrologic properties could be reasonably predicted from measurements of the volumetric water content held in rocks when dried in an oven at 60°C and 65 percent relative humidity. The minerals (generally zeolite) present in a rock that holds 5 percent water content under these drying conditions influence the in situ saturation (usually greater than 90 percent saturation) and saturated conductivity (reducing it by 2 to 3 orders of magnitude).

An important parameter that has not been characterized at Yucca Mountain is unsaturated hydraulic conductivity. Typically, this flow parameter is predicted from equations using moisture-retention data. An ultra-centrifuge has been configured at the Hydrologic Research Facility in Nevada Test Site Area 25 to measure unsaturated hydraulic conductivity directly under steady-state conditions. To date, two core samples have been tested. Moisture-retention curves also were measured, and the van Genuchten/Mualem equation (Van Genuchten, 1980) was used to predict conductivity, which was then compared with the measured data. Results indicated that the prediction of unsaturated hydraulic conductivity was very sensitive to the curve-fitting technique used to obtain parameters from the moisture-retention data. In addition, the unsaturated hydraulic conductivity was predicted most accurately when the estimated residual water content was not used in the data set. The technique somewhat under-predicted the measured values for the two samples of the pre-Pah Canyon bedded tuff (Tpbt2), suggesting that in modeling simulation, using constant flux boundaries, more water will flow through the matrix under unsaturated conditions than would be predicted by moisture-retention data alone. This interpretation is based solely on preliminary analysis of two samples; additional samples need to be measured before any definitive conclusions can be drawn.

Activity 8.3.1.2.2.3.2 - Site vertical borehole studies. The objectives of this activity are to define the distribution of pneumatic pressure, temperature, and water potential within the site unsaturated zone; and to determine in situ bulk-permeability and bulk-hydraulic properties of the combined fracture and matrix of the media within the site unsaturated zone.

Drilling

There was no site unsaturated zone drilling activity during the reporting period; this was an unfunded activity.

Vertical Seismic Profiling

No progress was made during the reporting period; this was an unfunded activity.

Borehole Instrumentation and Monitoring

Temperature monitoring in boreholes UE-25 UZ#4 and UE-25 UZ#5 in Pagany Wash indicates reversals in temperature-recovery trends at UE-25 UZ#5 in all instrument stations located above the Pah Canyon Tuff of the PTn (Table 3-2). After a prolonged cooling period, lasting almost 18 months, temperatures in the upper half of the PTn at UE-25 UZ#5 began increasing in November 1996. Borehole UE-25 UZ#5 is located on a hillslope adjacent to Pagany Wash. Temperature trends at UE-25 UZ#4, located in the channel bottom of Pagany Wash, continued to increase or were stable. No temperature reversals below the depth of penetration of surface-temperature changes have yet been noted in UE-25 UZ#4. The effect of the temperature data from the Pagany Wash site on estimates of net infiltration and deep percolation is discussed in Rousseau et al. (in prep.[a]). The recent temperature data from the two Pagany Wash boreholes may indicate the passage of an infiltration front through the Tiva Canyon Tuff and upper half of the PTn at a depth of about 70 m in UE-25 UZ#5.

Monitoring in borehole USW NRG-7a over the past six months indicated a steady recovery of temperature and water potentials in instrument station "D" located in the upper Yucca Mountain Tuff of the PTn. This recovery followed a one-month period of unusual oscillations and instability in the temperature and water potential measurements beginning on May 9, 1996 (see Progress Report #15, Section 3.1.7). The cause of the unusual behavior at USW NRG-7a is not known, but the data clearly indicate that the disturbance was propagated from above the instrument station and not from below. Topographic density-driven air-circulation within the Tiva Canyon Tuff offers one possible explanation to account for the extremes in the observed disturbances.

During January 1997, a leak test for instrument-station integrity was conducted in borehole USW SD-12 to confirm the reliability of pneumatic-pressure measurements from an instrument station located at the base of the lower nonlithophysal zone of the Topopah Spring Tuff. Before the onset of tunnel boring machine interference effects, the residual amplitude and phase lag of the synoptic pressure signal at this instrument station was larger than and led those of all other overlying instrument stations in the Topopah Spring Tuff. This observation has led some investigators to infer lateral (and preferential) communication between the Ghost Dance fault and borehole USW SD-12. The leak test indicated less than complete integrity for the instrument station in the lower nonlithophysal zone of the Topopah Spring Tuff but results were acceptable for all other instrument stations in USW SD-12. Thus, the data from the lower nonlithophysal instrument station cannot be used to infer preferential, lateral communication via the Ghost Dance fault.

The results of borehole monitoring are discussed further in the section below titled "Integrated Data Analysis and Interpretation."

Hydrologic Data Acquisition System

A basic data report was prepared and submitted to the Project Records Processing Center and Technical Data Base in December 1996. This report contains reduced pneumatic-pressure, temperature, and water-potential data for (a) boreholes USW NRG-7a, UE-25 UZ#4, UE-25

UZ#5, USW SD-12, and USW UZ-7a through September 30, 1996, and (b) borehole USW NRG-6 through September 12, 1996. Monitoring of USW NRG-6 was discontinued on September 12, 1996. A raw data package was submitted to the Project Records Processing Center and Technical Data Base in February 1997. This data package contained all raw data collected from August 15, 1996 through December 31, 1996, for the above-listed boreholes.

Surface-Based Air-Permeability Testing

No progress was made during the reporting period; this was an unfunded activity.

Integrated Data Analysis and Interpretation

Work began on a report that describes the results of surface-based monitoring in boreholes USW UZ-7a, USW SD-12, and USW SD-7 (located along the alignment of the ESF main drift), and USW NRG-7a (located near the intersection of the main drift and the ESF north ramp). The most significant findings contained in this report are summarized here.

Pneumatic pressures in all Topopah Spring Tuff instrument stations in boreholes USW SD-12, USW NRG-7a, and USW SD-7 have increased in response to ESF tunnel construction. These pressure increases are on the order of 0.06 kPa in USW SD-12, 0.09 kPa in USW NRG-7a, and 0.035 kPa in USW SD-7. At USW UZ-7a, only the bottom three instrument stations, located on the eastern side (foot wall) of the easternmost trace of the Ghost Dance fault, have shown an increase in mean pressure on the order of 0.020 kPa. Increases in mean pressure in the Topopah Spring Tuff reflect internal pressure adjustments in response to pneumatic bypassing of the overlying, impeding PTn by the ESF. The lack of any significant pressure changes in the Topopah Spring instrument stations located on the western side (hanging wall) of the Ghost Dance fault indicated very little attenuation of the atmospheric pressure signal and very high secondary porosity associated with fracturing in the fault zone.

Temperatures in all Topopah Spring Tuff instrument stations in USW SD-12 also have changed in response to ESF tunnel construction. Before the onset of tunnel interference effects, temperatures in these stations were increasing asymptotically toward their original, pre-disturbed, steady-state values. Following the onset of pneumatic-interference effects, temperatures have been steadily decreasing. These temperature reversals are viewed as an indication of heat loss driven by evaporation and unidirectional gas flow through the Topopah Spring Tuff near USW SD-12. Tunnel-induced temperature changes have not yet been observed in any of the other monitored boreholes at Yucca Mountain. Temperatures in the Topopah Spring Tuff instrument stations in USW NRG-7a, which is closer to the ESF tunnel than USW SD-12, have continued to increase even though pneumatic-interference effects in this borehole were first observed almost four months before those in USW SD-12. Data indicate that significant secondary porosity may be associated with the fracture system in USW SD-12. Indeed, the pneumatic-interference effects that occurred in USW SD-12 took at least two months to fully develop. As reported in Progress Report #15 (DOE, 1997e), pneumatic interference effects in USW SD-12 were initially very subtle and slowly increased over approximately one month. More recent analysis indicated that complete readjustment of the pressure system in USW SD-12 probably did not occur until sometime after April 1996—more than two months after the first indication that pressures had been disturbed. These data indicate that large volumes of gas were needed to reestablish pressure equilibrium in the vicinity of USW SD-12.

Analysis of the pressure records from the two deepest instrument stations in USW SD-12 (one located below the densely welded, vitric zone of the Topopah Spring Tuff and the other located at the top of the nonwelded Calico Hills Formation), indicate that the static pressure of these two stations is less than that predicted from extrapolating the static pressure profile developed across the Topopah Spring Tuff. The pressure "deficit" in these two stations is on the order of 0.12 kPa, which is much larger than the 0.001 kPa that can be accounted for by a change in the temperature gradient across the Topopah Spring Tuff-Calico Hills Formation contact. The pressure deficit may indicate an oxygen-consuming or reducing environment below the densely welded vitric subunit of the Topopah Spring Tuff. Pressure data from these two stations indicate the presence of a perched-water zone within the densely welded, vitric zone. A synoptic pressure signal is present in the pressure record of the station immediately below the perched-water zone. The amplitude of this signal is less than 1 percent of the surface signal but its phase lag with respect to the surface signal is essentially zero. This synoptic-signal component is superimposed on a seasonal pressure signal that represents a time-averaged composite of seasonal pressure changes occurring at the ground surface. The pressure signal from the underlying Calico Hills nonwelded unit contains no synoptic component, yet this signal leads the time-averaged composite signal in the overlying vitric zone. Taken together, these data indicate an atmospheric loading or strain-induced response associated with the synoptic signal component present in the pressure record of the instrument station below the perched-water zone. This finding indicates that the perched-water zone is of limited areal extent and accounts for the phase of the pressure signal in the deeper station leading the phase of the pressure signal in the overlying station. Furthermore, the synoptic component present in the pressure record of the overlying station represents the effect of local compression and expansion of the host rock that involves no net transport of gas.

Activity 8.3.1.2.2.3.3 - Solitario Canyon horizontal borehole study. The objectives of this activity are to determine the extent of fracturing, brecciation, and gouge development in the Solitario Canyon fault zone; to evaluate the effects of fault zone on ground-water movement in the unsaturated zone; and to identify additional fault-zone-related data needs.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: The systematic analysis of matrix-properties of the Topopah Spring Tuff middle nonlithophysal zone in the ESF will be extended to include saturated hydraulic conductivity for all samples. Additional samples will be collected in a more detailed transect with 140 samples at 20-m intervals. Fracture densities and apertures currently are being analyzed and will be used with matrix-property measurements to assess the likelihood of preferential pathways for flow, as well as to provide a data base of fracture properties for flow modeling. Additional unsaturated hydraulic-conductivity measurements will be made and compared with values predicted from moisture-retention data using several different techniques. Curve-fitting techniques also will be refined to predict measured data as accurately as possible.

Pneumatic-pressure, temperature, and water-potential monitoring will continue throughout FY 1997 in boreholes USW NRG-7a, UE-25 UZ#4, UE-25 UZ#5, USW UZ-7a, and USW SD-12. Data packages containing raw sensor readings and reduced values of monitoring data will be submitted to the Project Records Processing Center and Technical Data Base as appropriate. Borehole-monitoring data will be reduced and analyzed, including evaluations of barometric-pressure damping and lagging with depth, temperature gradients, and temperature stability. Emphasis will be placed on evaluating the significance of temperature changes induced by the tunnel boring machine in the Topopah Spring Tuff in USW SD-12 and transient temperature changes within the PTn beneath Pagany Wash in UE-25 UZ#4 and UE-25 UZ#5. A major effort during the next six months will be the compilation and writing of several subsections of the hydrology chapter of the Project Integrated Safety Assessment report.

3.1.8 Study 8.3.1.2.2.4 - Characterization of the Yucca Mountain Unsaturated Zone in the Exploratory Studies Facility

The objectives of this study are to supplement and complement the surface-based hydrologic information needed to characterize the Yucca Mountain site, and to provide information for analyzing fluid flow and the potential for radionuclide transport through unsaturated tuff.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tpcpul) are found in Buesch et al. (19[illegible]a).

Activity 8.3.1.2.2.4.1 - Intact-fracture test in the Exploratory Studies Facility. The objective of this activity is to evaluate fluid-flow and chemical transport properties of single, relatively undisturbed fractures. The purpose of the work is to characterize fluid flow along both undisturbed fractures and those under stress, and to provide these properties to help calibrate models for fluid flow in fractured rock at various scales.

No progress was made during the reporting period: this was an unfunded activity.

Activity 8.3.1.2.2.4.2 - Percolation tests in the Exploratory Studies Facility. The objective of this activity is to determine the hydrologic conditions that control the occurrence of fluid flow in fractured tuff units, and to provide experimental data against which the validity of conceptual and numerical models may be tested. In addition, this activity will determine the moisture balance within the ESF and adjacent rock mass in response to water-vapor transport from the ESF because of ventilation associated with ESF construction. This will be accomplished by measuring moisture conditions of the drift walls and rocks, and by monitoring the humidity and temperature of the air in the ESF.

Only the second part of this activity (moisture balance) was funded during the reporting period. This scope was added by Change Request 96/019.

Air temperature, relative humidity and wind speed data were collected in fixed locations within the ESF and on the tunnel boring machine. Three existing sensor stations are located near the north end of the ESF, and four new sensor stations were installed in the main drift and the south ramp as the tunnel boring machine advanced. Large variations in relative humidity and vapor density were observed, largely associated with variations in the mining operations, especially during weekdays when the ventilation system removes large amounts of construction water and moisture from the drift walls. During the weekends, when construction water was not used and ventilation was continued, the tunnel atmosphere stabilized to lower humidity conditions. Evaporation rates were also lower and variations were less. During weekends when the ventilation was continued, evaporation effectively increased the relative humidity to near saturated conditions. Rock temperatures near the tunnel boring machine were observed to change spatially and temporally and could be related to evaporation from the rock surfaces (Wang et al., 1996). The ventilation operations effectively remove the equivalent of approximately 200 mm/yr (± 100 mm/yr) of moisture. Therefore, these operations mask any possible measurements or observations of seepage into the ESF, which is estimated to be approximately one order of magnitude lower than the current ESF dryout rate (Bodvarsson and Bandurraga, 1996).

In addition, tensiometers and heat dissipation probes were installed in parts of the PTn in the south ramp to measure water potential. Time-domain reflectometry probes were installed in the Upper Paintbrush Contact Alcove to collect water-content data. The use of these "contact" sensors in specific locations has provided the opportunity to collect more accurate and more precise water-potential data than could be collected previously. Initial values of water potential measured in the PTn in the south ramp were -0.01 to -0.3 MPa, which are considerably "wetter" (with respect to a moisture-retention curve relating water potential to saturation) than any in situ measurements reported in the surface-based boreholes in similar zones of the PTn. These relatively greater water potentials suggest a higher flux through the PTn than would have been estimated using the surface-based borehole data. However, the sensors installed near the tunnel wall are drying out at a rate equivalent to 0.5 to 1 mm/day (180 to 365 mm/yr), which is supported by the dryout estimates made from changes in vapor density calculated using the temperature, relative humidity, and wind speed data. Matrix flux in the PTn has not yet been estimated using this new information.

Activity 8.3.1.2.2.4.3 - Bulk-permeability test in the Exploratory Studies Facility. The objectives of this activity are (a) to determine the scale at which the host rock behaves as an equivalent anisotropic porous medium, (b) to compare hydraulic test results against a distribution of simulated results calculated from a large number of realizations of the possible fracture networks conditioned on average fracture orientation and/or fracture density data, (c) to use a numerical fracture-flow model to establish the minimum dimensions at which other rock masses with the same fracture characteristics behave as equivalent porous media, and (d) to examine the dependence of rock-mass dimensions on changing saturation.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.2.2.4.4 - Radial borehole tests in the Exploratory Studies Facility. The objectives of this activity are to detect vertical movement of water in both the vapor and liquid phases, and to evaluate the potential for lateral movement of water along hydrologic contacts, as well as to evaluate the radial extent of excavation effects on the hydrologic properties of the unsaturated hydrogeologic units.

Project hydrologists completed and submitted a technical report to the Yucca Mountain Site Characterization Office (YMSCO) that documents pneumatic monitoring, hydrochemistry sampling, and air-injection and gaseous tracer testing conducted in the north ramp of the ESF between November 1994 and July 1996 (LeCain et al., in prep.). These studies were conducted in the Upper Tiva Canyon Alcove and the Bow Ridge Fault Alcove. Pneumatic monitoring of boreholes in the Upper Tiva Canyon Alcove showed no differential pressure among three boreholes, and between the boreholes and atmospheric pressure, confirming the large fracture permeability of the Tiva Canyon Tuff. In addition, gaseous-phase carbon-14 data indicate a large degree of mixing between atmospheric and rock gas. Matrix-properties testing of cores from boreholes in the Upper Tiva Canyon Alcove indicates that the matrix porosity of the Tiva Canyon Tuff crystal-poor, upper lithophysal zone (Tpcpul) ranges from 0.1 to 0.24, with a mean of 0.15. Matrix air-permeability values range from 1.4 to 120.0×10^{-17} m^2. Bulk permeability values of the Tpcpul from air-injection testing have an arithmetic mean of 28.6×10^{-12} m^2 and a geometric mean of 16.0×10^{-12} m^2. Comparison of the Tpcpul permeability values to the air-permeability values for the Tiva Canyon Tuff obtained form surface-based tests indicates that at shallow depths the Tiva Canyon Tuff is isotropic with respect to air permeability.

Activity 8.3.1.2.2.4.5 - Excavation effects test in the Exploratory Studies Facility. The objective of this activity is to monitor changes in both the stress state and fractured-rock permeability caused by excavating the ESF. The objective is to use these data, as well as other physical properties data gathered during the activity, to calibrate and validate a coupled hydraulic-mechanical, finite-element model. The model will be used to predict stress and ensuing permeability changes around excavation openings.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.2.2.4.6 - Calico Hills testing in the Exploratory Studies Facility. The objectives of this activity were intentionally deleted. Testing in the Calico Hills Formation will be described in revisions to other ESF studies.

This activity was deleted from the study plan in Revision 9 of the Site Characterization Program Baseline (DOE, 1995a). (See Appendix H for the Site Characterization Program Baseline history.) Testing in the Calico Hills nonwelded (CHn) hydrogeologic unit may be conducted as part of other ESF testing activities.

Activity 8.3.1.2.2.4.7 - Perched-water testing in the Exploratory Studies Facility. The objectives of this activity are to detect the occurrence of any perched-water zones; to estimate the hydraulic properties of the zones; and to determine the implication of the existence of such zones on the flux, flow paths, and travel times.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.2.2.4.8 - Hydrochemistry tests in the Exploratory Studies Facility. The objectives of this activity are to understand the gas-transport processes within the unsaturated zone and to provide independent evidence of flow direction, flux, and travel time of gas; to design and implement methods for extracting uncontaminated pore fluid from rock excavated during ramp construction; to determine the flow direction of water in the unsaturated zone by isotope geochemistry techniques; and to determine the extent of water-rock interaction so that geochemical modeling can be performed to deduce the flow path and to understand the geochemical evolution of unsaturated zone water.

In situ pneumatic monitoring and gaseous-phase chemical sampling were conducted in the Northern Ghost Dance Fault Alcove by temporarily instrumenting borehole NAD-GTB#1A with a 32.8-m, flexible, plastic borehole liner. The borehole liner has open ports attached to access tubing such that when the liner is everted into the borehole and inflated, it presses against the borehole walls, thus sealing the borehole and isolating the access ports. This allows discrete-interval pressure monitoring and gas sampling through the access tubing.

Results of gaseous-phase chemistry sampling indicate that essentially all the drilling air was removed successfully from 7 of the 10 sample intervals, as determined by the final concentrations of the SF-6 tracer gas. The three intervals that were not successfully purged of drilling air were near the far end of the hole where the tracer-gas injection system had malfunctioned.

Carbon dioxide concentrations in borehole NAD-GTB#1A ranged from 660 to 1180 ppm. The lowest concentrations were found in the three intervals closest to the access drift from which the hole was drilled, indicating that the rock gas in these intervals has a larger component of atmospheric air. This could be from better communication through the fault zone, better communication with the atmosphere, or communication with the air in the access and main drifts through the ESF walls. Carbon-13/carbon-12 ratios that ranged from -14.11‰ to -16.18‰ (parts per thousand) indicated a mixture of soil gas (-25‰) and atmosphere (-6‰), that is consistent with samples collected throughout the Yucca Mountain area. Gas samples were collected from borehole NAD-GTB#1A for carbon-14 age estimates, but the laboratory results are not yet available.

In other work, pore water extracted from rock samples collected from various locations in the ESF was analyzed for tritium. Although the presence of tritium in pore water does not allow the quantity of flux through the repository horizon to be determined, tritium is an indicator of the spatial distribution of flux. The presence of tritium on the order of 1 tritium unit (TU) indicates that the sample is partly composed of water that entered the ground after atmospheric testing of nuclear weapons started in 1952. Before 1952, background tritium concentrations in precipitation were about 2 to 4 TU. With a half life of 12.5 years, the remnant tritium activity in those waters would be less than 0.25 TU. Twenty-seven water samples for tritium analysis were distilled from (a) core collected during drilling of boreholes RBT#1 and RBT#4 in the Upper Paintbrush Tuff Contact Alcove in the north ramp, (b) core from borehole NAD-GTB#1A in the access drift of the Northern Ghost Dance Fault Alcove, (c) blast rubble obtained during the excavation of the Thermal Testing Facility, and (d) rock samples collected from "wet zones" in the ESF at Stations 7 + 57 (757 m) and 35 + 00 (3500 m).

At the 95 percent confidence interval (two standard deviations of analytical precision), the data indicate that tritium was present in only two samples, both from borehole RBT#4 in the Upper Paintbrush (Non-Welded) Contact Alcove. The presence of tritium near the Upper Paintbrush (Non-Welded) Contact Alcove is expected and is consistent with the concept that, at many locations, water percolates readily from the land surface to the top of the PTn.

However, given that percolating water would be subject to substantial mixing, particularly as it might move laterally within the PTn, any young water percolating downward from the PTn could be expected to have mixed with older water. The mixing would substantially reduce the tritium level in the mixture to an extremely small value. Therefore, consideration of a lower confidence interval of the data might be appropriate to interpret these very low values of tritium. At the 67.5 percent confidence interval, the data indicate that young water might have percolated below the PTn into the Topopah Spring Tuff middle nonlithophysal zone (repository horizon) at three additional locations in the ESF: (1) the Thermal Testing Facility, (2) the wet zone at Station 35 + 00 (3500 m), and (3) two intervals in borehole NAD-GTB#1A in the access drift of the Northern Ghost Dance Fault Alcove. Although at this level of confidence the presence of tritium from weapons testing in percolating water at these depths is somewhat uncertain, the tritium data do not preclude the possibility.

Activity 8.3.1.2.2.4.9 - Multipurpose-borehole testing. This activity was originally planned to monitor hydrologic and engineering interference effects from construction of exploratory shafts 1 and 2 on tests in these shafts and interference effects between tests in the shafts. In the current ESF design, with two ramps, testing in a scientific shaft is no longer planned.

This activity was deleted from the study plan in Revision 10 of the Site Characterization Program Baseline (see Appendix H).

Activity 8.3.1.2.2.4.10 - Hydrologic properties of major faults encountered in the main test level of the Exploratory Studies Facility. The objective of this activity is to investigate the permeability and flow conditions of the major faults encountered in ramps and drifts of the ESF.

Project hydrologists completed and submitted a technical report to YMSCO that documents pneumatic monitoring, hydrochemistry sampling, and air-injection and gaseous tracer testing conducted in the north ramp of the ESF between November 1994 and July 1996 (LeCain et al., in prep.). These studies were conducted in the Upper Tiva Canyon Alcove and the Bow Ridge Fault Alcove. Pneumatic monitoring in the Bow Ridge Fault Alcove indicated that the Tiva Canyon Tuff, Bow Ridge fault breccia, and pre-Rainier Mesa Tuff bedded tuff have high permeability. This conclusion is supported by gaseous-phase carbon-13 and carbon-14 isotopic data. Tritium values of matrix water indicate that within the last 50 years (since atmospheric testing of nuclear weapons), water has moved downward from the land surface along the Bow Ridge fault and mixed with older water resident in the pores of the rock matrix. Data from geothermal logging, however, did not indicate flow in the fault zone. Matrix-properties testing of cores from boreholes in the Bow Ridge Fault Alcove indicated that the porosity of the Tiva Canyon Tuff middle nonlithophysal zone is about 0.1, which is slightly higher than that of the

Tiva Canyon lower lithophysal zone. The matrix porosity of the Bow Ridge fault breccia ranges from 0.08 to 0.24, and the porosity of the pre-Rainier Mesa Tuff bedded tuff averages about 0.5.

Air-injection testing in the Bow Ridge Fault Alcove indicates mean bulk permeability values (arithmetic and geometric) of the Tiva Canyon Tuff middle nonlithophysal zone are 13.9 and 12.2×10^{-12} m^2, respectively, and for the Tiva Canyon Tuff lower lithophysal zone are 1.3 and 1.2×10^{-12} m^2. The three permeability values of the Bow Ridge fault breccia range from 8.0×10^{-12} m^2 to 15.8×10^{-12} m^2. The two permeability values of the pre-Rainier Mesa Tuff bedded tuff are 41.3×10^{-12} m^2 and 22.0×10^{-12} m^2. Cross-hole air-injection test results agree with the single-hole results and indicate that scale differences did not affect the test results. The pneumatic porosity estimate for cross-hole testing for the Bow Ridge fault breccia is 0.13; for the pre-Rainier Mesa Tuff bedded tuff, the estimates are 0.20 and 0.27. Comparison of the Tiva Canyon Tuff middle nonlithophysal permeability values in the ESF with values from surface-based tests indicates that the Tiva Canyon Tuff middle nonlithophysal zone is isotropic. However, comparison of the Tiva Canyon Tuff lower lithophysal permeability values in the ESF with the values from surface-based test indicates that the Tiva Canyon Tuff lower lithophysal zone is anisotropic and has a horizontal-to-vertical permeability ratio of approximately 10:1. Because water redistribution occurred during both the single-hole and cross-hole air injection tests, the capillary pressure of water held in fractures must be less than one atmosphere. Cross-hole gaseous tracer tests indicated effective porosities of 0.22 to 0.52 in the Bow Ridge fault breccia and 0.04 and 0.12 in the Tiva Canyon middle nonlithophysal and lower lithophysal zones, respectively. The tracer tests also indicated adsorption of the tracer in the fault breccia and tortuosity in the Tiva Canyon Tuff.

Borehole NAD-GTB#1A was drilled horizontally from the access drift of the Northern Ghost Dance Fault Alcove to a depth of 60 m to provide testing access to the fault before excavating the access drift through the fault. The entire borehole is located within the Topopah Spring Tuff middle nonlithophysal zone (repository horizon), but the character of the tuff in the 12-m wide fault zone has been altered. At this location, the fault zone includes four fault splays and is composed of a series of intervening clast-supported breccias and matrix supported, fine-grained breccias surrounded by relatively less fractured, welded tuff. Geothermal logging, conducted on November 7, 1996, identified a temperature drop across the fault zone. Following the geothermal logging, the alcove was excavated an additional 30 m reducing the length of NAD-GTB#1A to 30 m. Additional geothermal logging of the 30-m borehole on December 3, 1996 did not show the previously measured temperature drop across the fault zone but did indicate a small temperature increase at the main fault trace. Therefore, the temperature drop measured in November probably was the result of drilling-induced evaporative cooling that dissipated with time. The cause of the small temperature increase at the main trace measured in December is being investigated.

In situ pneumatic monitoring and gaseous-phase chemical sampling in borehole NAD-GTB#1A were accomplished as described in Activity 8.3.1.2.2.4.8. Pneumatic monitoring data indicated that different intervals within the 12-m fault zone have different pneumatic characteristics. All intervals showed attenuation and time lag of the barometric-pressure signal, although differences in lag time were almost imperceptible. Five of the seven monitored intervals in the fault zone, however, seem to have higher permeability values because their amplitudes are less attenuated. This conclusion is consistent with results of the air-injection permeability testing discussed in the following paragraph. Although the exact pathway of the barometric signal cannot be determined, possible pathways include (a) from land surface down the fault, (b) along the fault zone from the south where the main drift of the ESF intersects the fault, or (c) through the walls of the main drift along intersecting fractures. However, because the attenuation of the barometric-pressure signal is not sequential from the interval nearest to the main drift to the interval farthest from the main drift, the pathway for the barometric signal probably is not from the main drift. Rather, the varying response of the intervals to the barometric signal probably indicates different pathways within the fault zone.

Air-injection testing in borehole NAD-GTB#1A indicated that the permeability of the fault zone is more than an order of magnitude larger than the surrounding tuff. Air-injection testing was conducted on seven 1-m test intervals in the fault zone and five 1-m intervals outside the fault zone. Permeability values of the fault zone range from 1.3 to 11.1×10^{-12} m^2. Permeability values of the surrounding Topopah Spring Tuff middle nonlithophysal zone (repository level) range from 0.06 to 0.63×10^{-12} m^2. The average permeability value of the seven 1-m intervals tested in the fault zone is 5.5×10^{-12} m^2. Air-injection testing of a 12-m test interval that straddled the entire fault zone indicated a fault zone permeability value of 5.7×10^{-12} m^2. The average permeability value of the five 1-m test intervals outside the fault zone is 0.31×10^{-12} m^2. Water redistribution was identified in all five of the test intervals located outside the fault zone and in five of the seven test intervals located in the fault zone. The water redistribution pressures indicate that, although the capillary pressures are less than one atmosphere, they are highest near the main trace of the fault, indicating some drying near the main trace. The dryer zone may be the cause of the small temperature increase identified at the main trace of the Ghost Dance fault during the December 1996 geothermal logging.

Forecast: Collection of air temperature, relative humidity and wind speed data will continue in fixed locations in the ESF and on the tunnel boring machine. Analysis of ESF dryout will be enhanced by the reduction in water use that will occur when excavation of the south ramp is completed and the tunnel boring machine exits the south portal. Additional fixed-location temperature, relative-humidity, and wind-speed sensors will be installed in the south ramp as the tunnel boring machine progresses. The rock-wall instrumentation (heat dissipation probes, tensiometers, and two-domain reflectometry) will be incorporated into the PTn Lateral Diversion Study and the South Ramp Hydrology Study, although these instruments will continue to provide data for the dryout study. Several additional plans have been formulated to study the percolation processes along the main drift. Flow and evaporation testing and monitoring will be executed in short 5-m alcoves (niches) in a controlled environment. Longer borehole arrays (100 m) are planned to form an areal grid for the determination of percolation flux. Following completion of the ESF loop, combinations of monitoring tests in short and long boreholes and drifts will be used to better quantify the moisture balance in the ESF and to improve the assessment of hydrologic and pneumatic perturbations to the surrounding rock.

No work is planned for the second half of FY 1997 under the radial-boreholes testing activity.

The ESF hydrochemistry testing for the second half of FY 1997 will include distilling pore water for carbon-14 age estimates and tritium values from the Lower Paintbrush Tuff Contact Alcove and borehole NAD-GTB#1A in the Northern Ghost Dance Fault Alcove access drift. In situ pneumatic-pressure monitoring and gaseous-phase chemistry sampling will be conducted in the geothermal borehole to be drilled from the Southern Ghost Dance Fault Alcove access drift and in borehole RBT#1 in the Lower Paintbrush Contact Alcove.

Excavation of the Northern Ghost Dance Fault Alcove through the Ghost Dance fault and construction of a cross-hole testing facility will be completed. Testing of the Ghost Dance fault then will proceed with a sequence of geophysical and geothermal logging, pneumatic monitoring, gas sampling, and cross-hole air-injection and tracer testing. The testing will provide values of bulk permeability, pneumatic and effective porosity, and tracer travel times to support a three-dimensional, numerical model of the fault.

3.1.9 Study 8.3.1.2.2.5 - Diffusion Tests in the Exploratory Studies Facility

The objective of this study is to determine in situ the extent to which nonsorbing tracers diffuse into the water-filled pores of the tuffs of the Topopah Spring welded (TSw) unit at the main test level of the ESF. A diffusion test is also proposed in the Calico Hills nonwelded (CHn) unit.

No progress was made during the reporting period; this was an unfunded study. Work on this study has been suspended.

Forecast: No work is planned for this study during FY 1997.

3.1.10 Study 8.3.1.2.2.6 - Characterization of Gaseous-Phase Movement in the Unsaturated Zone

The objectives of this study are to (a) describe the pre-waste-emplacement, gas-flow field in the presence of open boreholes and the ESF excavations; (b) develop an understanding of the factors that produce and affect this flow field, including topographic, stratigraphic, and structural controls; (c) determine transmissive and storative properties for gaseous flow; (d) develop a history of air circulation at the instrumented boreholes from the time of drilling until the holes are stemmed, as an aid in evaluating the time following stemming before ambient conditions are restored; (e) determine fractured, porosity-gas-filled matrix porosity ratios and factors controlling gaseous exchange between the two, and/or the dispersivity of the fracture network to gas flow and transport, (f) determine changes caused in the gas-phase flow field as ESF excavations are advanced beyond open boreholes near the line of the ESF excavations; and (g) to develop a preliminary model of the transport of individual gaseous species.

Activity 8.3.1.2.2.6.1 - Gaseous-phase circulation study. The objectives of this activity focus on the collection and interpretation of several types of data, including composite borehole shut-in pressures, downhole gas-flow velocity surveys, gas-column temperature surveys, packed-off zone shut-in pressures, results from multiple borehole tracer tests, and gas chemistry and isotope chemistry from selected open boreholes and packed-off zones within boreholes. These data will be collected from a number of boreholes on an "as available" basis in order to collect as much data as possible in the boreholes most likely to be affected by ESF construction.

No progress was made during the reporting period. The work scope of this activity has been transferred to Activity 8.3.1.2.2.3.2 (see Section 3.1.7 of this progress report).

Activity 8.3.1.2.2.6.2 - Measurement of near-surface gas flow fields. The objective of this activity is to demonstrate how a water vapor RAMAN-LIDAR can be used to monitor and characterize preferred pneumatic pathways. The ability of this technology to measure the rate of water vapor exchange between the atmosphere and the preferred pathways will be validated and will provide benchmark data for convective transport models. This study will characterize known pneumatic pathways and locate the surficial expression of unknown pneumatic pathways at Devils Hole, Nevada using RAMAN-LIDAR. This information will aid the hydrology program by locating potential pneumatic pathways at Yucca Mountain.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for this study during FY 1997.

3.1.11 Study 8.3.1.2.2.7 - Hydrochemical Characterization of the Unsaturated Zone

The objectives of this study are to (a) characterize the hydrochemistry of the unsaturated zone by determining the transport mechanisms, flow directions, residence times, and travel times for gas and water; (b) determine the extent of water/rock interactions in the unsaturated zone; and (c) develop conceptual hydrologic and geochemical models.

Locations of most of the boreholes identified in the activities included in this study are shown in Figure 3-2.

Activity 8.3.1.2.2.7.1 - Gaseous-phase chemical investigations. The objectives of this activity are to understand gas-phase transport mechanisms within the unsaturated zone at Yucca Mountain, as well as to seek evidence of gas-flow direction, volume, rate, and travel time within the unsaturated zone.

Gaseous-phase carbon-isotopic data collected in May and July of 1996 from 15 stations ranging in depth from 24.7 to 435.9 m in instrumented borehole USW SD-12 were interpreted. The data generally show decreasing carbon-14 activities (increasing ages) with depth from 91.0 percent modern carbon (pmc) near the land surface to 24.4 pmc near the bottom of the borehole. These pmc values indicate apparent ages of about 800 years and 12,000 years respectively. As discussed in Yang et al. (in prep), these results for borehole USW SD-12 are very similar to those obtained in borehole USW UZ-1 for which it was concluded that the primary gas transport mechanism is downward diffusion of atmospheric gas. Two anomalously high carbon-14 activities of 88.7 pmc and 84.5 pmc at depths of 385.6 m and 435.9 m have been attributed to a leak in an instrument-station access tube and to contamination of rock gas by atmospheric air during drilling of the borehole. These conclusions were reached through detailed analysis of the time-series pneumatic-pressure data for the two stations (see Section 3.1.7, Activity 8.3.1.2.2.3.2 of this progress report).

Activity 8.3.1.2.2.7.2 - Aqueous-phase chemical investigations. The objectives of this activity are to design, test, and implement methods for pore-water extraction from core samples: to obtain hydrochemical data to evaluate ground-water flow direction, flux, and residence times in the unsaturated zone at Yucca Mountain; to evaluate the extent of water-rock chemical interactions; and to model the geochemical evolution of water in the unsaturated zone.

Recently collected tritium data are discussed in Section 3.1.8 of this progress report under Activity 8.3.1.2.2.8.

A report (Yang et al., in prep.) presenting a synthesis of new hydrochemical data obtained during FY 1996, as well as previously unreported data and the results of geochemical modeling was completed and submitted to the YMSCO. Unsaturated zone pore water in the PTn has significantly larger concentrations of major ions and dissolved solids than does perched water or saturated zone water. Recharge of perched or saturated zone waters, therefore, requires rapid flow through fractures or permeable regions in this unit to avoid mixing with the chemically concentrated water contained within the PTn. This conceptual model is consistent with observations of tritium and chlorine-36 in the deep unsaturated zone at Yucca Mountain. Further, occurrence of post-weapons-testing tritium in matrix water away from fracture zones indicates that some of the rapidly infiltrating water has spread laterally and down into nonwelded units.

Since delta deuterium (δD) values are larger than −99.8‰, most samples of unsaturated zone water and perched water are isotopically heavier than water from the last ice age, which has δD values of −101 to −103‰ and uncorrected carbon-14 ages between 12,000 and 18,000 years. If the matrix water in the Topopah Spring Tuff contained a significant amount of water from the last ice age, the δD values would be more negative. Therefore, pore water of the Topopah Spring Tuff has been interpreted to be of post-glacial origin (2,000 to 10,000 years old). In addition, gaseous-phase carbon-14 data from the Topopah Spring Tuff in borehole USW UZ-1 indicates ages between 2,000 and 10,000 years, which would be expected, assuming that the matrix water and the gaseous phase are in equilibrium.

Geochemical evolution of perched water, as calculated using the mass-balance model NETPATH, indicates the dissolution of volcanic glass in the tuffs at Yucca Mountain and growth of secondary minerals, such as clay (smectite) and zeolite (clinoptilolite) (Yang et al., in prep.). Modeling of the perched water in borehole USW UZ-1 indicates that the majority of dissolved sodium can be derived from two sources: (1) calcium-sodium and/or magnesium-sodium exchange, in which the amount of glass dissolution would be small, and (2) glass dissolution, in which sodium exchange would be small. More mineralogical information is needed from the unsaturated zone to determine which source is dominant. The composition of the perched water in borehole USW SD-9 indicates that significant quantities of glass dissolution and attendant precipitation of silica are required. For the perched water in borehole USW NRG-7a, dissolved sodium is derived principally from glass dissolution. NETPATH modeling was also used to convert carbon-14 ages for perched water based on the available carbon-13 and carbon-14 data for the mineral-phase carbon dioxide and calcite. The corrected carbon-14 residence times for the perched waters are 2150 to 2650 years for borehole USW NRG-7a; 5260 to 6260 years for borehole USW UZ-14; and 4040 to 5370 years for borehole USW SD-9.

Forecast: Gas samples for carbon dioxide concentrations and delta carbon-13 and carbon-14 analyses will be collected from the Calico Hills Formation in boreholes USW UZ-14 and USW SD-7 using a flexible borehole-liner system. The purpose of these studies is to determine whether or not the gaseous and aqueous phases are in hydrochemical equilibrium at this depth. Additional gas samples also will be collected from instrumented borehole USW SD-12 from those stations for which anomalous carbon-14 results were obtained during FY 1996 (see Activity 8.3.1.2.2.7.1).

The presence of tritium in the unsaturated zone will continue to be investigated by analyzing pore water extracted from rock cores. Cores will be collected from boreholes drilled in the Bow Ridge Fault Alcove, Lower Paintbrush Tuff Contact Alcove, and the Thermal Testing Facility in the ESF, and from boreholes drilled in both the Northern and Southern Ghost Dance Fault alcoves. Some additional tritium analysis will be conducted on pore water from cores of existing surfaced-based boreholes, including USW SD-7, USW SD-9, USW SD-12, and USW NRG-7a.

Carbon-14 residence times of pore water in the very-low-water-content, densely welded intervals of the Topopah Spring Tuff will be investigated by extracting carbon dioxide gas from the pore water using vacuum distillation. The carbon-14 activities of the pore water in the rock matrix will be compared with carbon-14 activities in fracture-derived perched water and fracture mineral coatings in the ESF. These data will be used to determine (a) the partitioning of unsaturated zone flow between the matrix and the fracture network and (b) the water flux through the matrix.

3.1.12 Study 8.3.1.2.2.8 - Fluid Flow in Unsaturated, Fractured Rock

The objective of this study is to develop and refine conceptual and numerical models describing both gas flow as well as liquid water and solute movement in unsaturated, fractured rock.

Activity 8.3.1.2.2.8.1 - Development of conceptual and numerical models of fluid flow in unsaturated fractured rock. The objective of this activity is to develop detailed conceptual and numerical models of fluid flow and transport within unsaturated, fractured rock at Yucca Mountain.

A memorandum report documenting a three-dimensional, fracture-network, flow model of the Topopah Spring Tuff using data obtained from the ESF was completed and submitted to the YMSCO (USGS, 1997a). The simulated network was calibrated by comparing simulated and mapped fracture intensities. Results showed that all mapped intensities fall within one standard deviation of the simulated intensities. Also, a visual comparison between simulated and mapped fractures showed that the simulated (representing only one realization) matches well with the mapped.

Preliminary modeling showed that there are three possible fracture sets: (1) striking N 65° W and dipping 85° SW, (2) striking N 3° W and dipping 85° W, and (3) striking N 33° E and dipping 85° NW. Fractures in set 1 are most abundant and fractures in set 3 are least abundant. The number of fracture connections between any two traceplanes in the simulation region, given that there is a connection, was sparse. The mean number of connections ranged from 1 for all scales to 5.7 for the 200-m scale. However, 80 percent of the realizations for the 200-m scale showed no connection from the south face to the north face of the box-shaped model domain. The number and pattern of connections showed little appreciable change with scale. The analysis calculated the number of fracture networks that connect a pathway to each of several traceplanes and the probability of connection.

Rock block analysis was used to help calculate an average rock block size used in dual-porosity simulations using the site-scale unsaturated zone flow model (see Section 3.1.13 of this progress report) to help evaluate potential imbibition surface area for small-scale problems and models.

Net directional permeability was calculated in the direction of gradient using Darcy's law, where the flux calculated from the finite-element analysis was based on assigned fracture-permeability distributions. Values ranged from 4.6×10^{-16} m^2 to 2.7×10^{-14} m^2.

Results of two-phase flow simulations showed that for different saturations, under a unit gradient, flux values ranged from 0.11 mm/yr out the bottom of the flow volume for no-flow boundaries on the edge faces, to 0.006 mm/yr out the bottom for variable-head boundary conditions. Given the in situ water-potential data collected in instrumented, surface-based boreholes (Rousseau et al., in prep.[b]), the flow simulation with no-flow boundaries on the edge faces seems to represent conditions in the Topopah Spring Tuff. For different saturations with no-flow boundaries on the sides, flux values varied by 2 orders of magnitude when comparing transmissivities calculated from low saturations to transmissivities calculated from high saturations. Therefore, the total flux leaving the block was about equally divided between the sides and the bottom when varying head conditions were used at the side boundaries.

Continuum properties of the discrete fracture flow simulation were evaluated to determine whether the network approached an equivalent continuum with respect to permeability. None of the scales tested (50-m, 100-m, 150-m, and 200-m) approached continuum properties, and only in a few instances did the flux values from opposite directions vary by less than a factor of two. From these results, it is doubtful that the fracture network in the tuffs, as analyzed, would ever approach continuum properties, because of the high degree of heterogeneity of the fracture system.

Activity 8.3.1.2.2.8.2 - Validation of conceptual and numerical models of fluid flow through unsaturated rock. The objective of this activity is to evaluate the reasonableness of the concepts on which the models developed under Activity 8.3.1.2.2.8.1 are based, by using the results of laboratory tests and tests performed in the ESF to access the adequacy of model performances.

No progress was made during the reporting period; this was an out-year activity.

Forecast: No additional work is planned or funded

3.1.13 Study 8.3.1.2.2.9 - Site Unsaturated Zone Modeling and Synthesis

The objectives of this study are to develop appropriate conceptual models for the site unsaturated zone hydrogeologic system; to select, modify, or develop numerical hydrologic models capable of simulating the hydrogeologic system and its component subsystems; to apply the models to predict the system response to changing external and internal conditions; to evaluate the accuracy of the models using stochastic modeling, conventional statistical analyses, and sensitivity analyses; and to integrate data and analyses to synthesize a comprehensive, qualitative, and quantitative description of the site unsaturated zone hydrogeologic system under present as well as probable, or possible, future conditions. For additional details, refer to the study plan.

Locations of most of the boreholes identified in the activities in this study are shown in Figure 3-2.

Activity 8.3.1.2.2.9.1 - Conceptualization of the unsaturated zone hydrogeologic system. The objectives of this activity are to develop conceptual models for the overall moisture flow system within the unsaturated zone at Yucca Mountain and to develop an internally consistent set of hypotheses that describes those aspects of the site hydrogeologic system that are needed to assess the capability of the site to isolate nuclear waste for a period of 10,000 years or longer.

Temperature profiles from the unsaturated zone were used to estimate the magnitude of the percolation flux at various locations at Yucca Mountain. This analysis recognizes that as water moves from cooler, shallower depths to warmer, deeper ones, heat is transferred from rock to the water percolating downward so that the downward moving water maintains thermal equilibrium with the surrounding rock. Therefore, the upward heat flow will decrease with increasing elevation along a borehole in a way that reflects the magnitude of the downward percolation flux. Because the decrease in heat flow depends only on the total mass of water moving through the rock, the percolation-flux estimates should reflect both the matrix and fracture components of the percolation flux, as long as the water and rock remain in thermal equilibrium. The analysis relies on estimates of the conductive heat flow at two elevations in a vertical borehole. The conductive heat flow is the product of the temperature gradient and thermal conductivity of the rock over the depth interval in which the temperature gradient is measured. Thus, percolation flux may be calculated as:

$$q_l = (q_{h1} - q_{h2})/\rho c \Delta T$$

where

q_l is the percolation flux in kg/s,

q_{h1} and q_{h2} are the conductive heat flows in W/m^2 at elevations z_1 and z_2 ($z_1 < z_2$),

ρ is the water density in kg/m^3,

c is the heat capacity of water (4.187×10^3 joules per kilogram per degree C), and

ΔT is the temperature difference between elevations z_1 and z_2, in degrees C.

As a first step, temperature gradients were determined for the Topopah Spring Tuff through least-square linear fit to the temperature data for that unit using data contained in Sass et al. (1988) and more recently collected data from instrumented boreholes (see Section 3.1.7 (Activity 8.3.1.2.2.3.2) of this progress report). Average thermal conductivity for the Topopah Spring Tuff was determined from data in Sass et al. (1988) to be 1.93 W/m/°C with a standard deviation of 0.29 W/m/°C. The heat flow at the lower elevation (q_{h1}) was taken to be either the conductive heat flow estimated for the saturated zone in the borehole, as given in Sass et al. (1988, Figure 17a), or the conductive heat flow determined for the Calico Hills Formation. The temperature difference was estimated using the temperatures in the middle of the Topopah Spring Tuff and either the water-table temperature or the temperature in the middle of the Calico Hills Formation, depending on where the heat flow at the lower elevation (z_1) was determined. Only a small subset of boreholes for which temperature data were given in Sass et al. (1988) had both unsaturated zone and saturated zone temperature data. Of these boreholes, only the temperature data from USW H-3, USW G-1, and USW G-3 lacked obvious evidence for the movement of heat through nonconductive processes, such as water movement within or along the borehole, or lateral water movement within the upper part of the saturated zone. Estimates of unsaturated zone percolation flux in these boreholes were 9.1 mm/yr in USW H-3, 23.5 mm/yr in USW G-1, and 4.9 mm/yr in USW G-3. Using temperature data from the Topopah Spring Tuff and Calico Hills Formation, estimates of percolation flux were 1.8 mm/yr in USW G-1 and 4.6 mm/yr in USW G-4. The difference in the two percolation flux estimates in USW G-1 (a) might be reflecting nonvertical flow in the lower part of the unsaturated zone, or (b) might be indicating that, even in the absence of obvious evidence in the temperature profile, nonconductive transport of heat in the saturated zone is introducing bias into estimates of the unsaturated zone percolation flux. If the latter is true, more reliable estimates of the percolation flux might be obtained if heat flows are determined using only data from the unsaturated zone.

A Project initiative titled "Unsaturated-Zone Model Expert Elicitation" began during the reporting period. The purpose of the initiative is to assess quantitatively the uncertainties associated with predictions of the spatial and temporal distribution of percolation flux by the site-scale, three-dimensional, unsaturated zone flow model. A series of three workshops to evaluate unsaturated zone flow characterization and modeling was conducted between November 1996 and January 1997. These workshops culminated in February 1997 in a series of interviews in which each expert estimated a probability distribution for percolation flux using data and interpretations presented in the workshops as well as additional analyses completed by each expert. The probability distributions were restricted to consideration of the spatially and temporally averaged percolation flux over the potential repository area. At each interview, individual experts also evaluated the reliability of dif   nt methods for estimating percolation flux from the reliability and availability of the neces... data, its sensitivity to percolation flux, and its treatment by various conceptual or numerical models.

Experts expressed variable amounts of confidence in any individual method for estimating percolation flux. In general, however, they endorsed the concept that the average percolation flux within the potential repository area could well be within the 5 to 10 mm/yr range and possibly higher. Their conclusions were based on the collective results from infiltration studies, analysis of temperature data, chloride mass-balance studies, percolation fluxes estimated during model calibration to saturation/water potential or isotope data (including carbon-14 and chlorine-36 data), and fracture-coating studies. Many of the experts felt that percolation rates beneath washes are larger than those presently depicted in infiltration maps, both because of surface runoff and as a result of shallow subsurface flow along the alluvium-bedrock contact. The experts felt that neither process was being captured adequately with the one-dimensional model used in the surface water-balance modeling. However, most agreed that significant net infiltration probably occurred only during years of above-average precipitation. They also generally believed that net infiltration probably was relatively low where alluvial cover was thick and vegetation had a greater opportunity to transpire infiltrated water, except where local conditions created focusing of runoff or subsurface flow.

Activity 8.3.1.2.2.9.2 - Selection, development, and testing of hydrologic-modeling computer codes. The objectives of this activity are to select, evaluate, and adapt existing numerical hydrologic-modeling codes for application to the site unsaturated zone hydrogeologic system; and to modify existing codes or develop new codes, as needed, to simulate particular problems or aspects that are unique to the Yucca Mountain system.

A new scheme for evaluating fracture-matrix interface areas in dual-permeability simulations was proposed and tested using the parameters in the unsaturat   zone flow model and the TOUGH2 code. The new scheme correlates effective fracture-matrix interface areas with upstream phase . aturation between fracture and matrix systems. The new treatment of the fracture-matrix interface will give physically reasonable approximations to flux calculations for dual-permeability simulations.

A modified form of the van Genuchten equation (van Genuchten, 1980) for capillary pressure has been added as an option in the TOUGH2 code. Linear extrapolation of the dry portion of the function gives greater (less negative) capillary pressures at low liquid saturations.

Activity 8.3.1.2.2.9.3 - Simulation of the natural hydrogeologic system. The objectives of this activity are to construct appropriate hydrologic models for the natural site hydrogeologic system to simu e and investigate the present state of the system, and to predict the probable future and past states of the system with regard to changes in environmental conditions.

A comprehensive comparison study investigated the differences in simulated mass fluxes using the equivalent continuum and dual permeability approaches with one-dimensional and three-dimensional models. The study results indicated that as long as the local equilibrium condition was reasonably satisfied the equivalent continuum model gave reasonable estimates of mass fluxes.

Spatially varying parameter distributions for porosity, thermal conductivity, rock grain density, and van Genuchten model (van Genuchten, 1980) fitting coefficients were incorporated into the unsaturated zone model (Banduragga et al., 1996). Matrix and fracture permeability and van Genuchten model-fitting coefficient distributions were developed using refined one-dimensional columns from the unsaturated zone model and by performing model inversions against available water potential and saturation data. The boreholes with core sample data included were USW UZ-14, UE-25 UZ#16, USW SD-7, USW SD-9, and USW SD-12. Other boreholes with borehole geophysical saturation data used in the inversions were UE-25 WT#12, UE-25 NRG#4, and UE-25 ONC#1. To provide matrix saturation estimates, the geophysical saturation estimates were corrected for the presence of lithophysal cavities in the Topopah Spring welded (TSw) hydrogeologic unit. The results of the inversions were interpolated to the three-dimensional model grid using geostatistical kriging techniques. Comparisons of modeling results using spatially varying parameters and parameters averaged over the layer were conducted using a two-dimensional vertical, south-north cross-section over the three-dimensional site-scale model domain. The surface boundary was subject to spatially varying infiltration using a mean of 4.9 mm/yr with a range of 0 to approximately 10 mm/yr, based on the infiltration data of Flint et al., in press. The value used for the repository block was 6.9 mm/yr. The simulated matrix liquid saturations from both models were compared with the observation data, which indicated that both models gave reasonable results.

A systematic study was performed to collect and analyze all the available fracture data from the ESF and boreholes. The purposes of this study were to determine fracture properties and their distribution in all the vertical formation layers and to provide fracture data input for the ITOUGH2 inverse modeling estimation of the hydrogeologic properties. An improved set of fracture parameters was developed for use in the unsaturated zone model using the recently available data on fracture geometries and spatial distributions from the ESF (D. L. Barr et al., 1996), borehole fracture frequencies and orientations ("Q"), and permeabilities from air-injection testing in boreholes (LeCain et al., in prep.). A computer code was developed to easily extract fracture statistics from the collected data on the basis of fracture size, orientation, and position in the ESF. The developed "Q" fracture property set based on ESF and borehole data and using air permeability (air-k) test results was incorporated into the unsaturated zone model. The set was used to rerun the combined column inversion (Bodvarrson and Banduragga, 1996) using the spatially varying infiltration map developed by the USGS (Flint et al., in prep.). The available pneumatic data were then used to constrain permeabilities in the PTn and TSw during the model inversions. This work was performed as part of the calibration of the gas model portion of the unsaturated zone model (Ahlers and Wu, 1997).

The development of a fracture property set for the unsaturated zone model continued, focusing on the calculation of van Genuchten parameters (van Genuchten, 1980) for fractures. The fracture hydrologic properties developed for the unsaturated zone model used air-injection permeability measurements performed at a range of spatial scales from surface borehole measurements (LeCain, 1997) as well as measurements obtained in the drift scale test area.

Calculations were performed using the permeability anisotropy and fracture frequencies in the Topopah Spring Tuff from the detailed ESF line survey data. Results were used to evaluate the ESF drift-scale tests. The most recent "Q" data from the ESF for fracture geometries were also incorporated (Wu et al., 1997b).

The three-dimensional site-scale model grid was modified to extend further to the north, 1 km beyond USW G-2, to include the possible higher water gradient effects. The grid was tested using two sets of parameters being developed. The steady-state simulations were performed using both the equivalent continuum model and the dual-permeability formulations, incorporating data from the latest infiltration map (Flint et al., in prep.).

Passive subsurface pneumatic pressure monitoring data were incorporated into calibration of the unsaturated zone model (Ahlers and Wu, 1997). Before this effort, calibrations of the unsaturated zone model had used either saturation plus water potential data or gas pressure data. By combining all three types of data into one calibration, better constraint of the model parameter set was achieved. The ITOUGH2 code was used to perform the inverse calibration. The inverse problem was very sensitive to the initial guess of the model parameter set. In particular, because of the initial guess, fracture parameters were not varied by ITOUGH2. The fracture parameters were restricted from field data, and the new parameters will be used as initial guesses in future calibrations.

A conceptual model for the perched water at Yucca Mountain was developed. This work is considering current isotopic data of waters (strontium-87/86, tritium, chlorine-36, and carbon-14 ages) and water chemistry (chloride) in light of the recent fracture geometry data from the ESF, distributions of infiltration, and relation to faults. Preliminary work was begun on submodels of the site-scale model to test various conceptual models in areas where chemical data from boreholes were obtained.

Development was completed of a near-surface source model for chloride for incorporation into the unsaturated zone site-scale model. This work, along with interpretations of chloride and other geochemical data, is described in Wu et al. (1997a). The geochemical and isotopic data from porewater and perched water are being used in the calibration of the unsaturated zone model. Two-dimensional dual permeability and three-dimensional equivalent continuum submodels were generated to model the chloride chemistry of Yucca Mountain, as well as to model environmental isotopes (chlorine-36) and radiogenic isotopes, such as strontium-87/86, that reflect water-rock interaction. Steady-state equivalent continuum model flow simulations were performed for the initial conditions for later transport simulations (Wu et al., 1997a).

The work performed on perched water can be summarized as follows:

1. The observed perched-water data from six boreholes were incorporated into a three-dimensional perched-water, unsaturated zone flow model. The model used a mean infiltration rate of 4.9 mm/yr, and an infiltration rate range of 0 to approximately 10 mm/yr. The infiltration rate value used for the repository block was 6.9 mm/yr. The vertical percolation flux rate calculated by this simulation was 4.9 mm/yr. The model simulated a stratigraphic diversion for ground water at the zeolitic layer below

the repository horizon. A slight diversion was also simulated at the PTn layer above the repository horizon.

2. The three-dimensional perched-water model was calibrated using observed perched water locations and moisture data from perched-water boreholes. Observed data were compared with the predicted perched-water locations, liquid saturation, and water potential data, and reasonable agreement was obtained The spatial distributions of perched-water bodies at Yucca Mountain were modeled.

3. Three simulated pumping tests were conducted in boreholes USW UZ-14 and USW G-2, and the simulated water levels of pumping and recovery periods were in good agreement with actual pumping data. From the pumping test analysis, the volumes of the perched-wa[illegible] bodies in USW UZ-14 and USW G-2 were estimated.

4. Perched-water ages were estimated using ground-water travel times through the fracture-matrix system, and the results are in reasonable agreement with residence ages determined from isotopic studies.

5. Historic high infiltration rates were simulated, and the impacts on perched-water level changes were predicted

6. The mean infiltration rate (4.9 mm/yr) used in the current three-dimensional site-scale model for simulating perched water is generally supported by analyses of average chloride content in the PTn. A model is being developed to simply incorporate chlorides into the unsaturated zone model; this model considers other geochemical data and isotopes.

The calibration efforts using the unsaturated zone model were completed (Wu et al., 1997b). The calibrations used all available data and inverse modeling techniques. The results of initial calibrations with one-dimensional models were put into the full three-dimensional unsaturated zone model for final calibration. Borehole temperature records were matched, resulting in best-estimate parameter sets for thermal conductivity Best-estimate permeability and van Genuchten parameters (van Genuchten, 1980) for both fractures and matrix were obtained by simultaneously matching saturation and water potential data from core samples and geophysical logging.

Forecast: In the conceptual modeling activity the temperature data will continue to be analyzed for the purpose of estimating percolation flux. Identifying suitable depth intervals in individual boreholes on which to base percolation-flux estimates will be emphasized. Also, an attempt will be made to determine the probability density function of percolation flux in a borehole, given the uncertainty in temperature gradients, in thermal-conductivity data, and in temperature differences that form the basis for the estimate.

During the next reporting period, the unsaturated zone flow modeling efforts will work on the three-dimensional permeability field conditioned to field data and information, calculation of infiltration at the drift scale resulting from episodic pulse infiltration, and modeling of future climate scenarios using the three-dimensional site-scale unsaturated zone flow model. Enhancement of the thermohydrologic modeling capabilities of the unsaturated zone will continue, as will ongoing support to the unsaturated zone flow model abstraction/sensitivity analysis process to the performance assessment activities supporting the viability assessment and License Application. A draft unsaturated zone flow model and a full report will be delivered as a Level 3 milestone summarizing unsaturated zone flow model development and simulations completed from September 1996 through May 1997.

3.1.14 Study 8.3.1.2.3.1 - Characterization of the Site Saturated Zone Ground-Water Flow System

The objectives of this study are to determine the internal and external boundary conditions and parameters that can be applied to the site saturated zone flow and transport models, and determine the rates and directions of ground-water flow.

Locations of most of the boreholes identified in the activities included in this study are shown in Figure 3-2.

Activity 8.3.1.2.3.1.1 - Solitario Canyon fault study in the saturated zone. The objectives of this activity are to characterize the hydrologic nature, significance, and implications of the Solitario Canyon fault, as well as to determine if the fault is a barrier to eastward flow of water in the saturated zone beneath the repository block.

No progress was made during the reporting period; this was an out-year activity.

Activity 8.3.1.2.3.1.2 - Site potentiometric-level evaluation. The objectives of this activity are to analyze the character and magnitudes of potentiometric-level fluctuations with depth and time to estimate transmissive and storage properties, to determine hydraulic gradient, to define altitude distribution of uppermost potentiometric surface, to determine long-term water-level trends, and to determine the water-level response to nearby pumping.

Monitoring continued of water levels in the saturated zone at Yucca Mountain. Quarterly or more frequent manual water-level measurements were made in 24 wells that monitored 31 depth intervals. Hourly water level data are no longer collected as part of the ground-water monitoring network. However, hydraulic and tracer testing at the C-hole complex was supported by recording hourly water levels in wells UE-25 WT#3, UE-25 WT#14, and UE-25 p#1, and in the upper and lower intervals of USW H-4. Continuous pumping of well UE-25 c#3 at approximately 150 gallons (568 L) per minute began on May 8, 1996 and continued through the end of the reporting period.

Manual water-level measurements for 1996 were reviewed and submitted to the Records Processing Center. Water levels in the Yucca Mountain site area remained stable during 1996.

A data report documenting ground-water levels at Yucca Mountain during 1994 (Graves et al., 1996) was published. A similar report on water levels during 1995 (Graves and Goemaat, in press) was prepared.

A report (Graves et al., in press) describing water-level trends at Yucca Mountain in the network of 28 wells and 36 depth intervals from 1985 through 1995 was approved for publication. The report indicates that mean annual water-level altitudes for all wells for 1985-1995 ranged from 727.93 to 1,034.60 m above sea level. The maximum change in water level over the 11-year period was 12.22 m in the lower interval of well USW H-3, and the minimum change was 0.31 m in the upper interval of well UE-25 b#1. In 31 of the 36 depth intervals monitored, the change in water-level was less than 1 m. No seasonal water-level trends were detected in any of the depth intervals monitored, and regional ground-water withdrawals did not appear to cause water-level changes. Most annual water-level fluctuations were attributed to responses to barometric-pressure changes and earth tides. Regional earthquakes, which occurred June 28-29, 1992, appear to have simultaneously affected the water levels in 7 of the depth intervals monitored. The maximum water-level change from the earthquake activity was in well UE-25 WT#6 where a rise in water level of 1.07 m was followed by a drop of 2.66 m. Well hydrographs over the 11 years of record where compared to determine if trends in water-level change could be related to wells completed in the same general area or with the same general water-table altitude. With the exception of wells USW WT-7 and USW WT-10, and to a lesser extent well USW VH-1, all of which are located in Crater Flat, no consistent, correlatable water-level changes are apparent in any two wells.

Continuous monitoring of the recovery data of the single-well aquifer test conducted April 8-25, 1996, in well USW G-2 ended on December 17, 1996. After 236 days of recovery, residual drawdown was 0.5 m. Analysis of drawdown and recovery data for the test indicate that fracture flow, dual-porosity flow, and boundary-effected flow occurred during the test. The Calico Hills Formation was the primary formation tested. Aquifer transmissivity was estimated to be 9 m^2 per day. The residual drawdown indicates that a perched-water body may have been permanently dewatered as a result of pumping. However, the impact of the potential perched water on the observed water level in well USW G-2 could not be determined with the data available from the test.

A report (O'Brien, in press) on the analysis of aquifer tests conducted in boreholes USW WT-10, UE-25 WT#12, and USW SD-7 was approved for publication. The single-well aquifer tests were conducted at the three boreholes between March 1995 and January 1996. The test in borehole USW WT-10, which was completed in the Topopah Spring Tuff, indicated a relatively high transmissivity of 1600 m^2 per day. Borehole UE-25 WT#12 was completed in the Topopah Spring Tuff and Calico Hills Formation and test results indicated a transmissivity of only 7 m^2 per day. The test conducted in borehole UE-25 WT#12 appears to have been significantly affected by well losses and apparently drew water from secondary fractures. Borehole USW SD-7, when tested, had been drilled into the base of the Calico Hills Formation about 4.5 m above the contact of the Prow Pass Tuff. The aquifer test conducted in borehole USW SD-7 was of a perched-water reservoir and test results indicated a transmissivity of 6 m^2 per day. The perched-water reservoir is approximately 150 m above the regional water table and had an estimated reservoir volume of 96,000 L at the time of the test.

Activity 8.3.1.2.3.1.3 - Analysis of single- and multiple-well hydraulic stress tests. The objectives of this activity are to determine intraborehole flow profiles for each well at the UE-25c (C-hole) well complex under static and pumping conditions, to correlate intraborehole flow rates with lithology and fractures, to identify test-scale hydrologic boundaries, and to estimate aquifer properties (e.g., transmissivity and storage coefficient).

This activity was completed. Results were reported in Geldon (1993), Geldon (1996), Geldon (1997) and Geldon et al. (in prep.).

Activity 8.3.1.2.3.1.4 - Multiple-well interference testing. The objectives of this activity are to discriminate between equivalent-porous-medium and fracture-network models at the scale of the tests, to determine effective aquifer properties (e.g., transmissivity and storage coefficient), and to evaluate the three-dimensional nature of the flow field in the test vicinity.

A long-term pumping test has been under way since May 8, 1996, in the Lower Bullfrog geohydrologic unit at the C-hole complex. The pumping well is UE-25 c#3 and the observation wells are UE-25 c#1 and UE-25 c#2 (at the C-hole complex), and UE-25 ONC#1, UE-25 WT#3, UE-25 WT#14, USW H-4, and UE-25 p#1 at locations ranging from 0.63 to 3.52 km from the pumped well. The pumping rate has been relatively constant at an average of 151 gpm (9.5 L/s). Superposed on this long-term hydraulic test are a sequence of shorter duration tracer tests (discussed under Activity 8.3.1.2.3.1.5).

Hydraulic connection between the C-holes and UE-25 ONC#1, which had been established during the May 1995 open-hole pumping test (Geldon et al., in prep.), was confirmed by the present test. Drawdown in UE-25 ONC#1 was detected about 140 minutes after pumping started and had reached about 0.91 feet (0.28 m) 379,000 minutes (37.6 weeks) after pumping started. In addition to a finite drawdown in UE-25 ONC#1, recirculating about 5 gpm (0.32 L/s) into UE-25 c#2 during tracer tests in May and October 1996 caused the water level in UE-25 ONC#1 (2,800 ft (853 m) away) to rise almost at the same time that the recirculation caused the water level to rise in UE-25 c#1 (about 280 ft (85.3 m) away). Furthermore, the cone of depression was elongated along an axis aligned in a west-north-west direction. The depression had been constructed using drawdown data from UE-25 c#1, UE-25 c#2, UE-25 ONC#1, UE-25 WT#14, UE-25 WT#3, and USW H-4 after 14,000 minutes (9.72 days) of pumping UE-25 c#3 during the open-hole pumping test in May 1995. Recent geologic mapping by the USGS (Day et al., in press) indicates possible discontinuous faults with associated fractures along a part of this alignment, which may be the reason for the hydraulic connections between the C-holes and both UE-25 ONC#1 and USW H-4.

When corrected for the effects of atmospheric-pressure changes, temporary pump shutdowns, tracer injection, and recirculation of water during tracer tests, the time-drawdown data from observation wells UE-25 c#1 and UE-25 c#2 up to 158,000 minutes (15.7 weeks) can be analyzed assuming either a single-porosity (confined, homogeneous, isotropic) or dual-porosity (fissure-block) aquifer. The Theis solution (Theis, 1935) for a single-porosity, confined aquifer produces a transmissivity value of 18,000 ft^2/day (1660 m^2/day), a hydraulic conductivity value of 90 ft/day (27 m/day), and a storativity value of 0.0002 for UE-25 c#1. The Theis solution produces a transmissivity value of 17,000 ft^2/day (1,620 m^2/day), a hydraulic conductivity value of 180 ft/day (54 m/day), and a storativity value of 0.001 for UE-25 c#2. For the same borehole and time period, the fissure-block solution tends to produce a lower value of transmissivity than the Theis solution. For UE-25 c#1, the fissure-block solution indicated a transmissivity of 12,000 ft^2/day (1150 m^2/day). The fissure-block solution separates hydraulic conductivity and storativity into fracture and matrix components. For UE-25 c#1 after 158,000 minutes, the fissure-block solution indicated fracture hydraulic conductivity of 60 ft/day (18 m/day), matrix hydraulic conductivity of 0.001 ft/day (0.0004 m/day), fracture storativity of 0.0003, and matrix storativity of 0.003. Transmissivity and storativity parameters are needed by performance assessment modeling of the saturated zone to quantify the flow field over which transport models are superposed to calculate radionuclide doses to the accessible environment.

After 158,000 minutes, the drawdown in UE-25 c#1, UE-25 c#2, UE-25 c#3, and UE-25 ONC#1 deviates from that predicted by either the single- or dual-porosity models, leading to speculation that either a barrier-boundary or a zone of reduced permeability in the Miocene tuffaceous rocks might have been reached.

Activity 8.3.1.2.3.1.5 - Testing of the C-hole sites with conservative tracers. The objectives of this activity are to determine aquifer transport properties, to evaluate applicability of equivalent-porous-medium models to analyze tracer tests, and to evaluate spatial correlation and scale-dependency of transport parameters.

Following establishment of a quasi-steady-state flow field by continuous pumping since May 8, 1996 (see Activity 8.3.1.2.3.1.4), a conservative-tracer test with a radially convergent flow field toward the pumped well, UE-25 c#3, was initiated on January 9, 1997. To begin the tracer test, 3.0181 kg of 3-Carbamoyl-2-Pyridone (Pyridone) and 11.35049 kg of 2,6 difluorobenzoic acid (2,6 DFBA), each dissolved in 210 gal (795 L) of UE-25 c#3 water, were injected into the Lower Bullfrog intervals of UE-25 c#1 and UE-25 c#2, respectively. Breakthrough of the 2,6 DFBA in the pumped well occurred approximately 5.07 days after injection and was measured at 13.6 ppb. The concentration reached a peak of 251 ppb approximately 14 days after injection and then decreased to the present value of 86 ppb approximately 41 days after injection. No breakthrough of Pyridone occurred during the reporting period.

Preliminary interpretation of the 2,6 DFBA test using the Moench (1995) analytic solution to the advection-dispersion equation for radially convergent tracer tests, produces a fracture porosity of 9.6 percent, a matrix porosity of 21 percent, and a longitudinal dispersivity value of 9 ft (2.74 m). The dimensionless Peclet number, which is defined as the ratio of longitudinal dispersivity to the interborehole distance, was 11. This analysis assumes a dual-porosity medium with primary transport of solute through discontinuous fractures connected by segments of matrix (total porosity of 9.6 percent), and a secondary process of matrix diffusion in which some of the tracer is stored in, and then released from, the pores of that part of the matrix not involved in the primary transport mechanism (porosity of 21 percent). Dispersivity and porosity parameters are needed by performance assessment transport modeling in the saturated zone to quantify the concentrations of radionuclides that would occur in the accessible environment as a result of a release from the proposed repository.

Fahy (1997) describes the analysis of a radially convergent conservative-tracer test in which iodide (as sodium iodide) was injected into the combined Lower Bullfrog-Upper Tram hydrogeologic interval in UE-25 c#2 while UE-25 c#3 was being pumped in February 1996. The paper describes results for the iodide test that are similar to those just summarized for the 2.6 DFBA tracer test.

Results similar to those obtained from the February 1996 iodide test and the January 1997 2.6 DFBA test also were obtained by analyzing the breakthrough curve of another tracer test in which pentafluorobenzoic acid was injected into the Lower Bullfrog interval of UE-25 c#2 in a partial-recirculation test in May 1996 (see Activity 8.3.1.2.3.1.7). All three of these tests were conducted using UE-25 c#2 as the injection well and UE-25 c#3 as the pumping well. The tests seem to produce the same transport parameters for the Lower Bullfrog-Upper Tram interval at the scale of the interborehole distance between these two wells, which is 95 ft (29 m).

Analysis of a tracer test in which iodide (as sodium iodide) was injected into the Lower Bullfrog interval of UE-25 c#1 in June 1996 while UE-25 c#3 was being pumped at 150 gpm (9.46 L/s) yielded a Peclet number of 12 (see Activity 8.3.1.2.3.1.7), similar to the value of 11 obtained from analyzing the three tracer tests between UE-25 c#2 and UE-25 c#3 described above. Because of the interborehole distance of 283 ft (86 m), however, this Peclet number represents a longitudinal dispersivity value of 18.33 ft (5.6 m) instead of the value of 9 ft (2.74 m) for an interborehole distance of 95 ft (29 m). The longitudinal dispersivity values of 2.74 and 5.6 m at interborehole scales of 29 and 86 m, respectively, seem to confirm the scale-dependence of this parameter, as postulated by Gelhar et al. (1992) and closely follow the pattern seen at the other field locations described.

A Windows-based personal computer program was developed to implement the Moench (1995) solution used for the analysis of the tracer tests just described (Umari, 1996). The program facilitates rapid experimentation with input parameters when attempting to match the theoretical Moench curves to the breakthrough curves obtained during a tracer test.

<u>Activity 8.3.1.2.3.1.6 - Well testing with conservative tracers throughout the site</u>. The objectives of this activity are to determine aquifer transport properties at selected site locations; to evaluate vertical and horizontal spatial variability of flow parameters; and to examine spatial correlation, cross correlation, and scale dependency of flow and transport parameters.

No progress was made during the reporting period; this was an out-year activity.

<u>Activity 8.3.1.2.3.1.7 - Testing of the C-hole sites with reactive tracers</u>. The objective of this activity is to characterize the chemical and physical properties of the geologic media in the saturated zone in the vicinity of the C-holes that will affect radionuclide retardation during ground water flow within the saturated zone.

Two conservative tracer tests that were initiated during FY 1996 [see Progress Report #15 (DOE, 1997e) for details] were completed. The first test involved injecting approximately 10 kg of pentafluorobenzoic acid into UE-25 c#2 on May 15, 1996, and the second test involved the injection of approximately 15 kg of sodium iodide (with iodide as the conservative tracer) into

UE-25 c#1 on June 18, 1996. In both instances, UE-25 c#3 was the production well These tests were conducted primarily to determine which well, UE-25 c#1 or UE-25 c#2, was better suited for injecting a reactive tracer and also to estimate of the mass of reactive tracer needed to conduct a successful test. The plan was to inject the reactive tracer, lithium ion, into whichever hole provided the highest peak concentration response of a conservative tracer because lithium recovery was expected to be lower and more delayed than that of a conservative tracer.

On the basis of the results of the two conservative tracer tests, a reactive tracer test was initiated on October 9, 1996. This test involved simultaneously injecting approximately 180 kg of lithium bromide (about 14.5 kg of lithium and 165.5 kg of bromide), 12 kg of pentafluorobenzoic acid, and about 7 g of 0.36 μm diameter polystyrene microspheres (about 3.5×10^{14} microspheres). The microsphere injection was initiated about 4 hours later than the solutes, but was completed at the same time as the solutes (i.e., a shorter injection duration). The bromide and pentafluorobenzoic acid served as conservative tracers with free diffusion coefficients that differed by about a factor of two. A comparison of the responses of these two tracers is expected to allow an estimate of the amount of matrix diffusion occurring in the system. The lithium response would then be compared with the response of the conservative solutes to estimate lithium sorption parameters in the system. The microspheres were intended to provide both an indication of the potential for colloidal contaminant transport in the system and to serve as a tracer that diffuses only very slowly, if at all, into the matrix.

All the tracer tests were conducted in an approximately 300 ft (91 m) packed-off interval in the lower Bullfrog Tuff extending from approximately 2300 to 2600 ft (701 to 792 m) below surface or 1000 to 1300 ft (305 to 396 m) below the water table at the C hole complex. This interval has the largest hydraulic conductivity of any major zone at the C holes. UE-25 c#3 was used as the production well in all the tests, with the production rate remaining nearly constant at ~150 gal (568 L) per minute. The distance between UE-25 c#2 and UE-25 c#3 at depth was approximately 30 m, while the distance between UE-25 c#1 and UE-25 c#3 at depth was about 80 m. All tests were conducted under partial recirculation conditions, with approximately 4 to 5 gpm (15 to 19 L/min) being recirculated from the UE-25 c#3 discharge into either UE-25 c#1 or UE-25 c#2 (about 4 gpm (15 L pmin) into UE-25 c#1 and about 5.3 gpm (20 L/min) into UE-25 c#2). The partial recirculation was established at least a day before the tracers were injected and was maintained in all instances for at least 16 days after injection. For the reactive tracer test initiated in October, recirculation was continued for 40 days.

Other than the different recirculation durations, the only significant difference between the reactive tracer test and the two conservative tracer tests was that a much larger volume of tracer solution was injected in the reactive tracer test (~12,000 L) than in the conservative tests (~1000 L). The larger volume was necessary in the reactive tracer test because it was desired to maintain approximately the same injection solution density in all tests (the conservative tests involved much smaller masses of tracers than the reactive test). A larger volume was also desirable to keep solute concentrations less than 0.2 M to avoid microsphere aggregation. Note that the volume of the packed-off borehole interval into which tracers were injected was about 4000 L in all tests.

The recoveries of each tracer as of January 30 (preliminary data) were ~56 percent for pentafluorobenzoic acid, ~56 percent for bromide, ~23 percent for lithium, and ~11 percent for microspheres. Each tracer breakthrough curve was bimodal (i.e., two peaks), and the tracer data exhibited clear matrix diffusion "signatures" (i.e., the pentafluorobenzoic acid peaks are higher than the bromide peaks because pentafluorobenzoic acid has less tendency to diffuse into the matrix than bromide, and the bromide tail crosses over the pentafluorobenzoic acid tail because of the greater mass of bromide diffusing from the matrix at late times). The lithium breakthrough curve (relative to the conservative tracers) showed clear evidence of diffusion and sorption in the matrix, as well as possible sorption in the flow pathways (assumed to be fractures).

The double-peak response in the reactive tracer test was markedly different than in the pentafluorobenzoic acid test initiated in May 1996. Project scientists attribute the different responses in the two tests to the different volumes of tracer solution injected, because all other variables were essentially the same in the two tests. Specifically the injection of approximately three packed-off interval volumes of tracer solution is believed to have resulted in tracers being forced into pathways that were not activated in the May test when only about one quarter of an interval volume was injected. The injection interval had no mixing or tracer distribution system, so possibly the dense tracer solution rapidly sank to the bottom of the interval in all tests. The first peak in October accounted for only about 12 percent of the mass of pentafluorobenzoic acid and bromide, so much of the mass injected in October may have still followed pathways that were activated in May. The June iodide injection into UE-25 c#1 resulted in only ~4 percent recovery of iodide, so UE-25 c#1 was eliminated from consideration for a reactive tracer test. All data discussed in this progress report should be considered preliminary until further quality checks and reviews are conducted.

The reactive tracer test data have been analyzed using a Laplace transform transfer function model to estimate transport parameters by simultaneously "fitting" the breakthrough curves of all solute tracers. The model assumes that the formation is a "dual-porosity" system in which flow occurs only in fractures, but the fractures are embedded in a porous matrix that contains a significant volume of stagnant water into which tracers can diffuse and sorb. The one-dimensional advection-dispersion equation is assumed to apply in the fractures, and diffusion into the matrix is modeled as a one-dimensional process occurring perpendicular to the direction of fracture flow. At least two sets of flow "pathways" had to be assumed to explain the bimodal breakthrough curves (i.e., two separate advection-dispersion equations were used to fit the data).

The interpretation procedure involved first simultaneously fitting the pentafluorobenzoic acid and bromide data by adjusting the following four parameters in each pathway, (1) the fraction of tracer following the pathway, (2) the mean fluid residence time, (3) the dispersivity, and (4) the mass transfer coefficient for matrix diffusion. The fits were constrained because all these parameters had to be the same for both pentafluorobenzoic acid and bromide because the two tracers were injected simultaneously and should have followed the same pathways in the same proportions. The fits were also constrained because the bromide diffusion coefficient in the matrix should have been about twice the pentafluorobenzoic acid diffusion coefficient (based on literature data). These constraints allowed the model parameters to be determined with much less uncertainty than would have been possible if only a single tracer breakthrough curve were analyzed.

The lithium response was "fitted" by adjusting lithium sorption parameters in both the fractures and the matrix under the constraint that the test- and formation-dependent parameters (i.e., the four parameters listed in the previous paragraph) had to be the same as those determined for the pentafluorobenzoic acid and bromide. Although the lithium fits are not completely unique, they strongly suggest that diffusion and sorption in the matrix occurs in all pathways and that sorption in fractures occurs in most pathways. The fits also suggest that there is good agreement between field- and laboratory-derived lithium sorption parameters. This agreement increases the confidence using laboratory-derived sorption data to predict field-scale transport behavior. Details of the interpretative analyses of the solute data were presented in Reimus and Turin (in prep.). The microsphere data have not yet been quantitatively interpreted.

A poster (Reimus, 1996) was presented at the Fall 1996 American Geophysical Union meeting in San Francisco, California, December 12-16, 1996. This poster summarized experimental work conducted during FY 1995 that showed that significant attenuation of polystyrene microspheres can occur over residence times of several hours in flow through porous media. The observed attenuation in a laboratory experiment may help explain the relatively low recovery of microspheres in the October reactive tracer test. The laboratory test results also indicated that smaller (~0.3 μm diameter) microspheres were less attenuated than larger (~1 μm diameter) microspheres, which suggests gravitational settling as a possible attenuation mechanism. This result prompted project scientists to use only small (~0.36 μm diameter) microspheres in the field tracer test.

Activity 8.3.1.2.3.1.8 - Well testing with reactive tracers throughout the site. The objective of this activity is to characterize chemical and physical properties of the geologic media in the saturated zone throughout the site that will affect radionuclide retardation during ground-water flow within the saturated zone.

No progress was made during the reporting period; this was an out-year activity.

Forecast: During the remainder of FY 1997, site water-level monitoring will include at least quarterly manual measurements in the saturated-zone network. Water-level data for 1996 will be reduced and organized into tables and hydrographs for publication. A report on the analysis of aquifer tests conducted in borehole USW G-2 during 1996 will be completed and submitted to DOE for concurrence and to the USGS Director for approval.

Conceptualization and analysis of drawdown after 158,000 minutes will be refined as drawdown data from UE-25 c#1, UE-25 c#2, UE-25 ONC#1 and other monitored observation wells continue to be evaluated.

In the multiple-well interference testing activity, pumping from well UE-25 c#3, and monitoring of wells UE-25 c#1, UE-25 c#2, UE-25 ONC-1, UE-25 WT#3, UE-25 WT#14, USW H-4, and UE-25 p#1, will continue until July 1, 1997. At that time, wells UE-25 c#2 and UE-25 c#3 will be reconfigured to allow hydraulic and tracer tests to be conducted in the Prow Pass hydrogeologic unit. Testing in the Prow Pass hydrogeologic unit will commence around September 1, 1997, and will provide estimates of hydraulic and transport parameters of a relatively low-transmissivity hydrogeologic interval. The Prow Pass hydrogeologic unit is important because it is one of the first horizons in the saturated zone that would be penetrated by radionuclides should they escape from a possible future repository at Yucca Mountain.

The Pyridone and DFBA conservative tracer test in the Lower Bullfrog began on January 9, 1997 and will continue until July 1, 1997. At that time, wells UE-25 c#2 and UE-25 c#3 will be reconfigured to allow hydraulic and tracer tests to be conducted in the Prow pass tuff, low-transmissivity interval. Testing in the Prow Pass will commence around September 1, 1997, and will provide estimates of hydraulic and transport parameters for this important low-flow interval.

Laboratory experiments will be conducted to assess the transport of lithium and other potential reactive tracers in artificially fractured C-hole cores and to measure the diffusion coefficients of various tracers used at the C-hole complex in relevant tuff matrices (from C-hole cores). In conjunction with these experiments, batch sorption and crushed tuff column transport experiments will be conducted to establish sorption parameters and to assess the applicability of these sorption parameters to fracture transport. The information derived from reactive tracer tests at the C-hole complex and from the supporting laboratory experiments will be integrated into the site-scale saturated zone transport model.

3.1.15 Study 8.3.1.2.3.2 - Characterization of the Saturated Zone Hydrochemistry

The objective of this study is to describe the composition of, and spatial compositional variations in, saturated zone ground waters using new and extant data; to identify the chemical and physical processes that influence ground-water chemistry; and to aid in the identification and quantification of fluxes to, from, and within the saturated zone.

Activity 8.3.1.2.3.2.1 - Assessment of saturated zone hydrochemical data availability and needs. The objectives of this activity are to compile and evaluate extant hydrochemical data for the saturated zone, to identify data deficiencies and potential sampling sites and assemble requisite material for sample and field data collection, and to augment extant information by collecting and analyzing new hydrochemical samples and data.

This activity has not been implemented. See Section A.1.1.3 of Appendix A of this progress report.

Activity 8.3.1.2.3.2.2 - Hydrochemical characterization of water in the upper part of the saturated zone. The objectives of this activity are to describe the hydrochemistry of the upper part of the saturated zone by collecting representative water samples from intervals within the upper 100 m of the saturated zone, within and adjacent to the site area, and studying their chemical and isotopic compositions; and to estimate flux to or from the saturated zone by collecting interstitial water and gas samples from immediately above the water table and studying their chemical and isotopic compositions.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.2.3.2.3 - Regional hydrochemical tests and analyses. The objective of this activity is to describe regional spatial variations in ground-water chemistry in the saturated zone by collecting representative water samples from wells and springs within the region and by studying their chemical and isotopic compositions.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.2.3.2.4 - Synthesis of saturated zone hydrochemistry. The objectives of this study are to describe the saturated zone hydrochemistry; to identify chemical and physical processes that influence ground-water chemistry; and to aid in the identification and/or quantification of ground-water travel times, climatic conditions during periods of recharge, flowpaths, and fluxes to, from, and within the saturated zone.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Selected existing saturated-zone hydrochemical data will be analyzed with a hydrochemical flow-path model to identify pertinent chemical processes and determine where in the saturated-zone flow system additional hydrochemical data would be most beneficial. New water samples will be obtained from well USW WT-17 plus two additional boreholes using techniques that ensure the representativeness of the samples. These samples will be analyzed for pH, Eh, major dissolved constituents, and isotopes of carbon, oxygen, hydrogen (including tritium), strontium, and uranium. Based on the results obtained from these new samples and the hydrochemical modeling, a sampling and analysis strategy for additional wells will be designed.

3.1.16 Study 8.3.1.2.3.3 - Saturated Zone Hydrologic System Synthesis and Modeling

The objectives of this study are to synthesize the available data into a model and make a qualitative analysis of how the system is functioning and to represent quantitative observations of hydrogeologic data pertaining to the ground-water flow system in a comprehensive flow model.

Locations of most boreholes identified in the activities making up this study are shown in Figure 3-2.

Activity 8.3.1.2.3.3.1 - Conceptualization of saturated zone flow models within the boundaries of the accessible environment. The objectives of this study are to synthesize the available hydrogeologic data to develop a conceptual model and to make a qualitative analysis of how the site saturated zone hydrogeologic system is functioning. This activity includes development and calibration of the numerical, site-scale, saturated zone flow model prior to its application in Activity 8.3.1.2.3.3.3.

A report describing the conceptual model of the saturated zone flow system (Luckey et al., 1996) was published.

Work continued on updating the preliminary, site hydrogeologic framework model. The updates included refining the grid-cell size from 1500 m to 250 m, including cross-sectional data from the Nevada Test Site, and including the upper surface of the Paleozoic carbonate aquifer along the east-west seismic line from Crater Flat to Jackass Flats as interpreted by Hunter et al. (1996).

Problems with mesh-generation software used during previous model scoping runs were identified and corrected. A new mesh was generated and tested for use with the ground-water simulation code FEHMN, and model-input data arrays were constructed. Potentiometric data for the area were evaluated, screened, and incorporated into the new mesh. Over 100 potentiometric data points were identified for use in the model for comparison against simulated hydraulic pressures (hydraulic heads). Alternate meshes also were generated that contain alternate conceptual models for the potentiometric surface in the Yucca Mountain area. Scoping runs were made on preliminary meshes using the parameter-estimation software, PEST. To provide faster simulation times, the model domain was re-evaluated and scaled down to a rectangular box 30 km wide, 45 km long, and 1.5 km thick surrounding Yucca Mountain. Calibration simulations began toward the end of the reporting period.

Work resumed on analyzing water-level fluctuations to estimate aquifer hydraulic characteristics. Data from 11 zones in 6 wells have been evaluated and analyzed. Preliminary results indicate that, although the method was developed for porous-media aquifers rather than the fractured-rock aquifers that dominate flow at Yucca Mountain, the method provides values of transmissivity and storativity that are consistent with values previously obtained using other aquifer-test methods. A comprehensive table of hydraulic-characteristic data for the Yucca Mountain area was compiled.

Work began on writing a synthesis report that documents the results (through April 30, 1997) of the site-scale, saturated zone, flow modeling. An annotated outline for the report was prepared and reviewed.

Work began on planning a workshop to discuss the abstraction and testing of saturated zone process models (flow and transport) for use in performance assessment modeling.

<u>Activity 8.3.1.2.3.3.2 - Development of fracture network model</u>. The objectives of this activity are to relate results of hydraulic and conservative tracer tests in wells to fracture-network flow characteristics at Yucca Mountain, to develop methods to analyze hydrologic data to determine fractured rock flow characteristics, and to model flow in the saturated zone in the vicinity of the C-hole complex.

<u>Confirm Saturated Zone Hydrologic Flow Models</u>. In recognition of the need for a better understanding of saturated zone flow at the scale of Yucca Mountain to support the waste containment and isolation strategy, this new sub-activity was developed and funded for the first time in FY 1997. The objectives of this activity are to develop analytic capabilities to design and analyze saturated zone hydraulic tests in the immediate Yucca Mountain site area and to test hypotheses concerning ground-water flow in the saturated zone in the site area to support the development and application of the site-scale, three-dimensional ground-water flow model that is

being constructed to support the total system performance assessments for viability assessment and license application.

To meet the objectives, a sub-site-scale, three-dimensional, numerical, ground-water flow model of the saturated zone hydrologic system is being developed. The model domain covers approximately 100 km^2 (12 km x 8 km area) that encompasses the area beneath the potential repository, the large hydraulic gradient zone, the Solitario Canyon fault, Fortymile Wash, and the area immediately down-gradient from the potential repository. The model is being developed as an analytical tool to test hypotheses of saturated zone ground-water flow in the immediate site area and to analyze the results obtained from aquifer tests at the C-hole complex and at other sites. The model is intended to simulate hydrologic conditions and processes at a small scale such that it can be applied in designing and analyzing results from the planned second saturated zone test complex. All available geologic, hydrologic, and hydrochemical data are being used to develop, calibrate, and test the model.

A plan for developing the subsite-scale model was submitted in October 1996 (Cohen et al., 1996). The plan defined the model boundaries and discussed the vertical layering scheme. Also discussed in the plan were the grid scheme, boundary conditions, and initial conditions, and treatment of features of interest (e.g., large hydraulic gradient, Solitario Canyon fault, C-hole complex) by finer discretization of the mesh in these areas. The plan discussed data needs for model parameters and data limitations, if known. Features or processes included in the model, and upon which sensitivity analyses will be performed, include temperature, hydrochemistry, potential upwelling from carbonates, and the role of faults.

The three-dimensional subsite-scale model grid was constructed and fault properties and characteristics thoroughly assessed to determine which faults to include in the model. Hydrochemical data were analyzed at the subsite-scale and compared with major ion and isotope patterns found in data at the regional scale. Hydrochemical signatures were contoured at two depths in the lower volcanic aquifer, which is included in the Crater Flat hydrogeologic unit (CFu) (Table 3-2). [Note: The "lower volcanic aquifer" is a collective reference to the combination of the Prow Pass Tuff, the Bullfrog Tuff, and the Tram Tuff of the Crater Flat Group (Table 3-2). Data about these units are being collected by the C-hole testing.] The contours delineated areas of recharge and showed a distinct change from higher to lower concentrations of many ions across the Solitario Canyon fault. The contours, however, did not discern local flow domains because of the paucity of saturated zone hydrochemical data near Yucca Mountain, as compared with the regional scale. Cohen and Simmons (1997) reviewed and evaluated relevant saturated zone features and processes at the subsite-scale (including temperature, upflow along faults from thermal convection, hydrochemistry) and cursorily examined mixing between layers. The report provided initial insights at the subsite-scale for predicting flow down-gradient from the repository.

<u>Activity 8.3.1.2.3.3.3 - Calculation of flow paths, fluxes, and velocities within the saturated zone to the accessible environment</u>. The objectives of this activity are to estimate ground-water flow direction and magnitude for input into travel-time calculations; and to evaluate the porous-media concept and fracture-network concept for determining flow paths, fluxes, and velocities.

No progress was made during the reporting period; this was an out-year activity.

Forecast: The sub-site-scale three-dimensional numerical model of saturated zone flow will be delivered in August 1997 and will include the numerical model (input and output data sets) and a report describing the model, the degree of model calibration, and model uses and limitations. The report will examine data from the C-hole complex and will include recommendations for design of the second saturated zone test complex. The model will be used as a tool to help determine the optimum location of the test and test configuration, and to predict results of the hydrologic flow and transport tests to be conducted. A progress report on the model will be completed in May 1997 that will identify any unexpected results or difficulties encountered.

The site hydrogeologic framework model will be reviewed and approved for release to the Records Processing Center. Water-level fluctuations will be analyzed to estimate hydraulic characteristics of the aquifers and submitted for review. The site flow model will be calibrated to the extent possible by April 30, 1997. A synthesis report describing the preliminary site flow model, with model results to April 30, 1997, will be completed and submitted for review and approval. After April 30, 1997, work will begin on refining the calibration of the site flow model. A workshop to discuss the abstraction and testing of the site flow and transport models, for use in performance assessment modeling, will be held and recommendations concerning the abstraction and testing processes will be made

3.1.17 Related International Hydrological Work

No progress occurred during the reporting period. As of November 8, 1995, the subsidiary agreements under which the cooperative work had been conducted were terminated and all international collaboration was discontinued.

The Office of Civilian Radioactive Waste Management (OCRWM) had bilateral agreements with Canada (Atomic Energy of Canada Limited [AECL]), Switzerland (Swiss National Cooperative for the Storage of Radioactive Waste [Nagra]), and Sweden (Swedish Nuclear Fuel and Waste Management Company [SKB]) and has participated in activities of international organizations such as the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA), the European Commission (EC), and the International Atomic Energy Agency (IAEA).

3.2 GEOCHEMISTRY (SCP SECTION 8.3.1.3)

The changes to the Geochemistry Program since the SCP was issued are summarized in Appendix A, Section A 1.2

3.2.1 Study 8.3.1.3.1.1 - Ground-Water Chemistry Model

The objective of this study is to develop a ground-water chemistry model that will initially describe pre-emplacement conditions. Future changes in these properties and processes will then be considered, including infiltration changes as influenced by climatic conditions; long-term mineralogic changes, particularly those influenced by the thermal pulse from emplaced waste; and material property changes caused by the emplaced waste, or possible igneous activity.

Work focused on integrating the conceptual models of controls on ground-water chemistry (discussed above) into the mountain-scale transport model. The impact of evapotranspiration on chloride concentrations in waters has been incorporated and soil-zone precipitation reactions, such as calcite and/or silica precipitation, are being incorporated into the transport model

Forecast: Work will continue on incorporating conceptual models for controls on ground-water chemistry into transport models. In addition, field work designed to measure the oxidation-reduction potential in waters from several saturated zone wells will be planned.

3.2.2 Study 8.3.1.3.2.1 - Mineralogy, Petrology, and Chemistry of Transport Pathways

The objectives of this study are to determine the three-dimensional distribution of mineral types, compositions, abundances, and petrographic textures within the potential host rock; and to determine the three-dimensional distribution of mineral types, composition, and abundances in rocks beyond the host rock that provide pathways to the accessible environment.

Activity 8.3.1.3.2.1.1 - Petrologic stratigraphy of the Topopah Spring Tuff. The objective of this activity is to determine the petrologic variability within the devitrified Topopah Spring Tuff at Yucca Mountain and to define the stratigraphic distribution of variability.

Core samples from drillholes along the path of the ESF were being analyzed to determine mineralogic stratigraphy in the densely welded, devitrified Topopah Spring Tuff. Analytical data on quantitative mineralogy were being collected from drillholes USW NRG-7, USW SD-12, and USW SD-7, representing the north-to-south run of the main drift of the ESF. Particular attention was being given to the zonation with depth of the major silica minerals (tridymite, cristobalite, and quartz) because of (a) the impact that high cristobalite contents have on determining the need for respiratory protection and (b) the importance of silica mineral type in estimating silica solubility in waters released under thermal loading of a repository. Preliminary data from USW NRG-6 core suggest that ratios of quartz to tridymite plus cristobalite can be used to quantify lithophysal character within the chemically homogeneous host rock.

Activity 8.3.1.3.2.1.2 - Mineral distributions between the host rock and the accessible environment. The objective of this activity is to determine the three-dimensional distribution, chemistry, and total abundance of all major rock-matrix minerals, between the host rock and the accessible environment. The analysis of the three-dimensional stratigraphy will be most heavily weighted toward those units that will first be encountered along potential flow paths away from the repository (i.e., Calico Hills Formation).

A report on the three dimensional mineralogic model of Yucca Mountain (Chipera et al., 1997) describes the accompanying computer model, including a description of the data sources, software, assumptions, derived mineral volumes, and the model limitations. The report also discusses the magnitude of expected increased model uncertainty if non-"Q" data are excluded. The model, in its preliminary form, focuses on mineral volumes and distributions important to site performance. Minerals and mineral groups described include smectite + illite, the sorptive zeolites, analcime, cristobalite + opal-CT, and tridymite. Distributions of glass were also included in the model. The model included all available quantitative x-ray powder diffraction data from the surface down to the Paleozoic basement. The model has been produced in a format that is amenable for use in process-level flow and transport models using the FEHM and TOUGH2 computer codes. The geologic and lithologic stratigraphy used in the model is consistent, to the extent possible, with the Reference Information Base Section 1.12(a). Stratigraphy

The three-dimensional mineralogic model is currently being used in developing abstraction methodologies to treat variability in site mineralogic properties. A proposal for testing to address the effects of mineralogy and heterogeneity on radionuclide transport was developed for the workshop on unsaturated zone radionuclide transport, held in February 1997 (see Section 3.2.5 of this progress report).

<u>Activity 8.3.1.3.2.1.3 - Fracture mineralogy</u>. The objective of this activity is to determine the distribution of minerals within fractures at Yucca Mountain, within all significant rock masses that might provide transport pathways with some component of fracture flow.

Recent work (Vaniman and Chipera, 1996) shows that the trace-element geochemistry of calcites at Yucca Mountain provides important information about transport processes in fractures. Fracture calcite occurs in both saturated and unsaturated hydrologic zones in the tuffs at Yucca Mountain, Nevada. In the upper unsaturated zone, the major constituents of the calcite crystal structure (carbon, oxygen) originate at the land surface. At greater depth there is a "barren zone," straddling the water table, where calcite is rare and mixing of surface and subsurface sources may occur. Deep in the saturated zone, distinctive manganese calcites reflect deep sources, including calcium released as analcime and albite formed or carbonates derived from underlying Paleozoic rocks. In the unsaturated zone and in the barren zone, above the deep manganese calcites, variations in calcite lanthanide chemistry can be used to distinguish rhyolitic from quartz latitic sources. Lanthanide ratios and strontium contents of calcites record the chemical evolution of waters flowing through the unsaturated zone and upper saturated zone. Variations in calcite chemistry in the unsaturated zone and in the barren zone show that (a) strontium, which is readily exchanged with clays or zeolites, is essentially removed from some flowpaths that are in contact with these minerals and (b) traces of manganese oxides found in the tuffs significantly affect ground-water chemistry in the unsaturated zone and in the barren zone by removing almost all cerium from solution (evidenced by characteristic cerium depletions in calcite throughout this zone). Extreme cerium removal may be a result of cerium oxidation (cerium+3 or cerium+4) at the surfaces of some manganese oxides, particularly rancieite. Higher strontium contents and lack of cerium depletions in the deeper manganese calcites reflect different ages, origins, and transport systems. The calcite record of lanthanide and strontium transport in the unsaturated zone shows that minor minerals (clays and zeolites) and even trace minerals (manganese oxides) will affect the compositions of ground waters that flow over distances greater than a few tens of meters.

A suite of sample splits was collected for mineralogic and geochemical analysis based on the results of isotopic analyses, particularly of calcite carbon-14 ages, strontium-87/strontium-86 ratios, and carbon-13 values. The goals of this parallel sample analysis, with splits collected for mineralogic and trace-element analysis taken from well-characterized generations of calcite, include determining

1. The geochemical significance of early versus late mineralization. (Has the geochemical environment of calcite deposition changed over time? Did some calcite precipitate from early, warm aqueous systems during the cooling of the tuffs?)

2. The stratigraphic controls on calcite trace-element chemistry. (Is the model of Vaniman and Chipera (1996) valid, indicating the validity of using natural geochemical tracers to infer effective transport distances for sorption within fractures?)

3. Alternative possible explanations for the geochemical anomalies in calcite compositions at Exile Hill.

4. Variations in calcite composition that correlate with zones where chlorine-36 from weapons testing is evident.

5. Whether absence of cerium anomalies at or above repository depth indicate pathways that bypass the upper vitrophyre of the Topopah Spring Tuff.

Preliminary results show that early-formed calcites may be significantly restricted in lanthanide-element contents, suggesting one specific depositional environment that communicated widely throughout the site, possibly operating soon after tuff emplacement. In addition to the splits of calcite and opal collected additional materials are being analyzed, either of other mineralogies (clays, manganese-oxides) or from other localities. The other localities include other stations in the ESF and drill cores (USW NRG-6, USW NRG-7, USW SD-9, and USW SD-12) that lie near the path of the ESF. These additional samples will be used to place the data to be obtained from the ESF samples in vertical stratigraphic context.

Forecast: Additional data will be incorporated into the three-dimensional mineralogic model of Yucca Mountain as more analyses of core become available. Comparative geostatistical and mineralogic model sections will be prepared. The results will be used to support the total system performance assessment for the viability assessment. Further studies involving microautoradiography, combined with fracture-flow studies and traditional batch sorption studies, will be used to obtain the information needed to determine the impact of trace minerals, such as smectite, for modeling the retardation of radionuclide transport. Staff will continue to analyze and interpret on textural and geochemical bases the origin of calcite, opal, manganese oxide, and clay minerals from the ESF. Samples from cores collected from boreholes near the ESF (especially USW SD-7, USW SD-9, and USW SD-12) will also be analyzed to provide information on mineral variability with stratigraphic level. This information is needed to interpret the data from the ESF horizon. Mineralogic analyses are being obtained for an additional 175 borehole samples, including specific intervals of USW NRG-7/7a, USW UZ-14, USW SD-7, USW SD-9, and USW SD-12 (with supporting data from USW H-3) to fill gaps in the three-dimensional mineralogic model of Yucca Mountain. Particular emphasis will be placed on the distributions of minerals important in transport and repository thermal loading analysis and, in the repository horizon, the abundances and distributions of regulated hazardous minerals (e.g., crystalline silica). The three-dimensional mineralogic model will be used in meeting the needs of performance assessment in unsaturated zone radionuclide transport modeling, particularly in developing the testing of abstraction methodologies.

3.2.3 Study 8.3.1.3.2.2 - History of Mineralogic and Geochemical Alteration of Yucca Mountain

The objectives of this study are to determine the timing, temperatures, and hydrologic conditions of past alteration at Yucca Mountain; and to study experimentally the dehydration of smectite, zeolites, and glass. Processes range from deep-seated past hydrothermal alteration to ongoing shallow mineral deposition along fractures and faults.

Activity 8.3.1.3.2.2.1 - History of mineralogic and geochemical alteration of Yucca Mountain. The objectives of this activity are to constrain the timing, geochemical transport, and paleohydrology of hydrothermal, digenetic, and epigenetic alteration; to estimate the long-term thermochemical stabilities of important sorptive phases, such as clinoptilolite; and to investigate the natural evolution of phases such as silica polymorphs that can influence water composition, rock hydrologic properties, and the stabilities of other silicate minerals.

Research focused on correlating mineralogy and alteration in the PTn hydrogeologic unit with measured hydrologic properties of the unit. Mineralogic characterization of the same samples that have been used for hydrologic-property measurements makes it possible to evaluate differences in moisture content as determined by different methods because these differences are likely to correlate with hydrous mineral content. Beyond this immediate use, progress was made on developing a conceptual model of alteration in the PTn that will be used to predict the distribution of hydrologic properties within the unit using mineralogic and hydrologic property variations resulting from alteration processes.

A milestone progress report (Levy and Chipera, in prep.) provided descriptions and examples of natural alteration that have been identified and studied. Examples were found of both rootless hydrothermal alteration related to the cooling of the pre-Tiva Canyon tuffs or of the overlying Tiva Canyon Tuff and ambient-condition digenetic alteration. One hypothesis of alteration to be tested is that ongoing argillic alteration is occurring in the pre-Tiva Canyon tuffs and that localization of this alteration reflects differences in surface infiltration above the pre-Tiva Canyon tuffs. The report included a discussion of criteria for evaluating this hypothesis, as well as pertinent preliminary data from the ESF south ramp and drillholes USW UZN-31 and USW UZN-32.

Another aspect of the correlation of mineralogic alteration and hydrologic properties is the use of mineralogic alteration to identify fractures and faults within the PTn that have hosted fluid flow. Cooling-stage alteration of the pre-Tiva Canyon tuffs is of particular interest as an indication of possible changes to be expected from repository-induced alteration of this unit. Products of alteration associated with cooling probably included quartz/chalcedony, opal-CT, opal-A, heulandite-clinoptilolite, and smectite. Several processes related to mineralogic alteration probably altered the hydrologic properties of the rock. These processes include localized devitrification along fracture boundaries, fracture spallation (with alteration of fracture aperture), pore cementation, and selective genesis or deposition of smectite along fractures and faults.

Information about the transmissive pathways through the pre-Tiva Canyon tuffs is required to help interpret the chlorine-36 isotopic data from the ESF and drillholes. Mineralogic and textural characteristics of faults and fractures cutting the pre-Tiva Canyon tuffs were studied. It is uncertain whether the examples examined to date represent bounding or typical attributes of faults in the pre-Tiva Canyon tuffs. Minor faulting or fracturing in minimally consolidated vitric nonwelded pre-Tiva Canyon tuffs produced very little breakage of rock grains, but this was apparently sufficient in some instances to allow fluid transport that resulted in localized mineral deposits.

Activity 8.3.1.3.2.2.2 - Smectite, zeolite, manganese minerals, and glass dehydration and transformation. The objectives of this activity are to determine how minerals and glasses important in the rocks at Yucca Mountain will dehydrate and transform under expected thermal loads; and to investigate the ability of zeolites and smectites to rehydrate after the peak in temperature.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: This study will continue to correlate variations in alteration with existing and new data on hydrologic properties and to identify needs for additional measurements. The work will require analysis of samples from the ESF, especially from a series of short drillholes in the north ramp PTn exposures, and from unanalyzed drill cores. Assumptions about mineral distribution and field-scale mineralogic and textural features (e.g., vitric-zeolitic transition, syngenetic alteration zones) will be evaluated. The evaluations and documentation will incorporate ESF data on the abundance of transmissive features identified by mineralogic and textural studies to help define and support selection of input terms for infiltration into the TSw. This effort will be closely integrated with the studies of hydrologic properties by performing mineralogic-petrologic analysis of hydrologic-properties samples. The study will also be integrated with chlorine-36 isotopic studies of the water movement test (e.g., Study 8.3.1.2.2.2; see Section 3.1.6 of this progress report).

3.2.4 Study 8.3.1.3.3.1 - Natural Analog of Hydrothermal Systems in Tuff

The objectives of the study are to improve the reliability of long-term predictions regarding hydrothermal rock alteration in devitrified welded ash-flow tuff; to test the capabilities of the EQ3/6 geochemical code through modeling of alteration mineral assemblages in natural systems; and to provide a better understanding of the origin of alteration mineral assemblages currently found in Yucca Mountain.

As part of an effort to consolidate work scopes, this study plan will not be written because much of the work overlaps activities in Study Plans 8.3.4.2.4.1 and 8.3.1.20.1.1 (discussed in Sections 5.2.2 and 3.14, respectively, of this progress report). Much of the work is a compilation effort to select analog site(s) combined with testing and/or calibration of the EQ3/6 code to reduce uncertainties in EQ3/6 applications.

No progress was made during the reporting period; this was an unfunded study. Work scheduled in support of this study has been suspended.

Forecast: No activity is forecast for this study.

3.2.5 Study 8.3.1.3.3.2 - Kinetics and Thermodynamics of Mineral Evolution

The objectives of this study have been revised to determine experimentally the kinetics and thermodynamics of mineral alteration at Yucca Mountain and to produce a model for past and future mineral alteration at Yucca Mountain. The model is intended to explain the natural mineral evolution resulting from the transformation of metastable mineral assemblages to more stable assemblages and to predict the possible effects of emplacement of radioactive waste in a repository.

The revised study plan for Kinetics and Thermodynamics of Mineral Evolution at Yucca Mountain was accepted by the Project on December 11, 1996. This study now combines previous Studies 8.3.1.3.3.2 and 8.3.1.3.3.3 and will investigate the thermodynamics of clinoptilolite, mordenite, analcime, silica minerals, and, to a lesser extent (because of their much smaller abundance), smectite and illite.

A report (Carey and Bish, in prep.) describes the enthalpy of hydration of natural clinoptilolite as determined by isothermal immersion calorimetry. The data in this paper represent the first direct measurements of the energetic consequences of hydration and dehydration reactions that can occur at Yucca Mountain. The measurements were made on natural clinoptilolite that was cation exchanged to produce calcium-, sodium-, and potassium-end members. Heats of immersion of clinoptilolite were determined and compared with the enthalpy of hydration determinations from a thermogravimetric study on the same samples (Carey and Bish, 1996) are similar to but of smaller magnitude than the values of enthalpy of hydration. Because of the data collection and analysis methods used, the heats of immersion data are believed to be more accurate and more precise because of the lower uncertainty in the measurements. The effects of dehydration of clinoptilolite on the thermal evolution of the potential repository at Yucca Mountain were considered by comparing the amount of energy consumed by dehydration with the energy necessary to heat rocks lacking hydrous minerals. The energy consumed on heating clinoptilolite from 25 to 200°C is between 70 and 80 percent above that required for nondehydrating materials. These results indicate that accurate thermohydrologic-hydrologic modeling of rock units at Yucca Mountain should consider the thermal effect of dehydration-hydration processes in clinoptilolite and other hydrous minerals, in addition to the water produced or adsorbed during heating or cooling.

Bish (1996) emphasized the variety of ways in which mineralogy may be important in the long-term isolation of radioactive wastes, including the importance of minerals in retarding the migration of actinides and other radionuclides that form large, complex aqueous species. Minor and trace minerals, such as smectite and iron and manganese oxides and hydroxides, are particularly important in retarding the migration of complex actinide species such as neptunyl.

Carey et al. (1996) described the effect of cation exchange and dehydration on clinoptilolite-analcime equilibria as applied to conditions at Yucca Mountain. Primary calorimetric data for clinoptilolite and analcime were extended by a modified thermodynamic estimation procedure and combined with recent cation-exchange (Pabalan, 1994) and hydration (Carey and Bish, 1996) studies to yield the thermodynamic properties of arbitrary compositions of clinoptilolite and analcime. The equilibrium breakdown of clinoptilolite to analcime was calculated assuming cation-exchange equilibrium with aqueous solutions of variable sodium to calcium and sodium to potassium ratios, as functions of temperature and silica activity. The results show that the strong affinity of clinoptilolite for potassium or calcium over sodium limits the stability region of analcime to silica-poor, sodium-rich solutions. The clinoptilolite stability field is maximized for aluminous compositions in equilibrium with siliceous analcime. Increased temperatures decrease the stability field of clinoptilolite, but this effect is moderated slightly by the loss of water from clinoptilolite. Estimates of uncertainty in these results were derived by comparison of predicted and measured calorimetric data for clinoptilolite and analcime. The uncertainties are significant with respect to the predicted stability field of clinoptilolite.

An analysis has been completed of the parameters affecting calculations of the kinetic evolution of minerals. The analysis is an integrated summary of the relation between theoretical and experimental kinetics and the implementation of numerical evaluations of kinetic evolution at Yucca Mountain. One difficulty in this implementation is that, unlike the experiments, the natural environment has poorly constrained surface areas for reacting minerals and poorly understood effective diffusion constants for aqueous species. Laboratory experiments used samples with measured surface areas exposed to a vigorously stirred aqueous solution. The natural system consists of minerals with unmeasured surface areas that are incompletely exposed to an aqueous solution having concentration gradients over distances that have not been characterized.

The magnitude of the uncertainty for the reactive surface areas of minerals in the natural environment is a function of the surface roughness and the relative exposure of the mineral grains to pore fluids. Surface roughness increases the reactive surface area by a factor as large as 1×10^1 relative to the geometric surface area (e.g., the surface area of a sphere). The relative exposure of a mineral grain to pore fluids may be bounded by considering the difference between minerals exposed only along a fracture and close-packed spherical minerals. The exposed surface area differs by 1×10^3 in this instance. Considering an average situation, the estimated surface areas of reactive minerals in the natural environment (including Yucca Mountain) have an uncertainty of 1×10^2 to 1×10^3 which translates directly into an uncertainty of 1×10^2 to 1×10^3 in the mineral reaction rates.

An analysis was completed of the reaction mechanisms of the silica polymorphs at Yucca Mountain. Several contrasting views of silica polymorph evolution have been outlined (Duffy, 1993). These include processes governed by:

1. dissolution and precipitation rate constants as exemplified by the experimental work of Rimstidt and Barnes (1980);

2. an Ostwald ripening process in which the polymorphs evolve in a step-wise process from opal-A to opal-CT to quartz as suggested by Murata et al. (1977);

3. a defect-concentration model proposed by Duffy (1993); or

4. diffusion-limited reaction of the silica polymorphs (application of process 1 in a nearly closed-system environment). All these models have been evaluated with respect to the observed distribution of silica polymorphs at Yucca Mountain. These natural data show that nucleation kinetics are not significant because quartz is present throughout the stratigraphic column. However, opal-CT/cristobalite disappears at depth in a manner that is perhaps consistent with either process 3 or 4.

Staff participated in the workshop on unsaturated-zone radionuclide transport in Albuquerque, New Mexico, February 5-7, 1997. At this workshop, an abstraction-testing draft proposal was formulated for sensitivity studies to examine the effects of mineral alteration on unsaturated zone radionuclide transport. Results suggest that repository-induced alteration of the existing minerals and glasses in Yucca Mountain tuffs may change the hydrologic properties (permeability and porosity) and geochemical properties (sorption capacity, composition including H_2O) of the natural media over time. In addition, the workshop determined that simulations of radionuclide transport in the unsaturated zone should consider the possible changes in rock properties.

Forecast: The updated conceptual model of mineral evolution will be completed. This update will integrate information for mineral evolution at Yucca Mountain using kinetic, thermodynamic, field, and analytical studies of mineral, glass, and water reactions. Data will be compiled from other studies and from the literature, as appropriate. This activity will define and bound important mineral reactions in the host rock that may affect transport of radionuclides. This activity will support transport model development and calculations, as well as near-field studies. The conceptual model will be incorporated into a transport and thermohydrologic model such as FEHM, and a sensitivity analysis will be conducted of the effects of individual mineral reactions on radionuclide transport. Information obtained in these studies will be combined with the FY 1997 preliminary three-dimensional mineralogic model for Yucca Mountain (Chipera et al., 1997) to produce an updated conceptual model of mineral evolution within a mineralogic framework that is an improved representation of Yucca Mountain. The effort will summarize the potential geochemical and mineralogical effects of thermal loading and the possible impacts on radionuclide containment and rock properties. This work will culminate in a model of the expected behavior of the secondary phases at Yucca Mountain as functions of time, temperature, water and mineral composition, and water-vapor pressure.

Abstractions for the total system performance assessment will continue through this fiscal year in an attempt to include in the assessment the effects of mineral alteration on unsaturated zone radionuclide transport. Staff will use existing Project thermohydrologic models to determine temperature and saturation as a function of time and depth at various locations throughout the repository block. Using this information, potential locations of mineral and glass reaction will be predicted. The sensitivity analysis may couple data on mineral alteration with the thermohydrologic models to model mineral alteration effects more explicitly. This procedure is expected to develop a history of the evolution of properties in the unsaturated zone for use with unsaturated zone radionuclide transport modeling to test the effects of predicted mineral alteration on transport. The thermohydrologic model will be used as input to the conceptual model of mineral alteration. The product will consist of a sensitivity analysis using a coupled mineralogic-thermal chemical model (one or two-dimensional) to determine if thermal mineral alteration should be considered in the total system performance assessment. If so, staff will provide an appropriate abstraction that represents the future properties of the unsaturated zone flow and transport regime below the repository horizon.

3.2.6 Study 8.3.1.3.3.3 - Conceptual Model of Mineral Evolution

The objectives of this study are to formulate a model to explain the observed distributions of minerals in Yucca Mountain. The evolution of framework silicates (feldspars, zeolites, and silica polymorphs) will be emphasized. The model will also address the general chemical evolution of vitric tuffs and will be used to predict future mineral evolution in the mountain caused by both natural processes and development and operation of a repository.

This study has been combined with Study 8.3.1.3.3.2; see Section 3.2.5 of this progress report. Forecast for the combined study is reported in Section 3.2.5 of this progress report.

3.2.7 Study 8.3.1.3.4.1 - Batch Sorption Studies

The objective of this study is to obtain sorption coefficients for key radionuclides as a function of important geochemical parameters. These studies will statistically evaluate the experimental results and will provide the data base that will be used to develop models to predict sorption coefficients under conditions not directly addressed by the experimental program.

Activity 8.3.1.3.4.1.1 - Batch sorption measurements as a function of solid phase composition. The objective of this activity is to determine sorption coefficients for radionuclides on the zeolitic and vitric tuffs of the Calico Hills nonwelded hydrogeologic unit, on devitrified tuffs, and on pure minerals representative of the minerals present in the rock and fractures of the repository block.

Project geochemists have initiated studies to validate the sorption data base for whole rocks. Microautoradiography studies of selected radionuclides and lithologies that are important to performance have begun. Radionuclides for these studies are uranium, plutonium, and americium; rock types include vitric, zeolitic, and devitrified tuffs.

Activity 8.3.1.3.4.1.2 - Sorption as a function of sorbing element concentrations (isotherms). The objective of this activity is to characterize the dependence of sorption coefficients upon the aqueous concentration of the element being sorbed by developing isotherms for the radionuclides. This activity will develop isotherms for the radionuclides to be tested. These isotherm data will be incorporated into the sorption data base for use in determining element concentration levels at which precipitation begins to contribute to the measured sorption ratio, and in modeling sorption to predict retardation along flow paths.

No progress was made during the reporting period; this was an out-year activity.

Activity 8.3.1.3.4.1.3 - Sorption as a function of ground-water composition. The objective of this activity is to measure sorption coefficients as a function of ground-water compositions anticipated along potential travel paths. These data will contribute to the sorption data base and support the sorption model development and performance assessment calculations.

Software was written to ensure accurate and efficient transfer of experimental data to the sorption data bases. This software also provides data analysis and statistical interpretation capabilities. The sorption data bases have been formally submitted to the Project. These data bases provide fast access to and selection of data needed to evaluate the effects of various parameters on sorption. The data bases contain information on the solid and water used, the test atmosphere, temperatures, radionuclide concentrations, experiment durations, and whether experiments were sorption or desorption experiments. All data needed to calculate the K_d are also incorporated as are data necessary to provide traceability to the original data sheets. The statistical uncertainty in the K_d arising from analytical uncertainties is calculated and presented. K_ds are calculated using both the initial radionuclide concentration of the experiment and the final concentration in a parallel control experiment done without solid present. These K_d values are used along with uncertainties to assess acceptability of the data. The data bases provide an accessible, easily searchable, and easily sortable resource. This resource can be used to develop sorption isotherms or solid phase mineralogy to investigate the effect of parameters, such as radionuclide concentration, by sorting for mineral composition and comparing the K_d variation within a given mineralogy to the variation among different mineralogies. Thus, appropriate K_ds can either be selected for transport modeling or a determination can be made that additional data are needed.

Activity 8.3.1.3.4.1.4 - Sorption on particulates and colloids. The objective of this activity is to determine if sorption of important radionuclides occurs on particulates or colloids that may be present in ground waters along potential transport pathways. Batch techniques, modified to accommodate the much smaller sample sizes, will be used to measure sorption. If any sorption is measured, the use of sorption coefficients alone may not accurately predict the transport of sorbed radionuclides.

Project scientists initiated a set of batch sorption experiments to study the reversibility of radionuclide sorption onto colloids. The experimental matrix consists of two types of colloids (clays and iron oxides), bicarbonate ground waters (UE-25 J#13 and UE-25 p#1), and two chemical forms of plutonium (polymeric and soluble).

Activity 8.3.1.3.4.1.5 - Statistical analysis of sorption data. The objective of this activity is to produce statistical correlations and error estimates. Various statistical approaches will be used on the sorption data to determine those variables (e.g., mineralogy, ground-water composition, and atmosphere) that most profoundly affect the sorption coefficients; to predict sorption coefficients as a function of mineralogy and, perhaps, ground-water composition; to estimate errors associated with predicted sorption coefficients; and to identify gaps in the experimental data.

No progress was made during the reporting period; this was an out-year activity.

Forecast: This study will integrate the radionuclide solubility, sorption, and transport synthesis reports submitted in FY 1996 to identify the most technically valid models that should be used to describe these processes in the performance assessment. Sorption data will be collected on plutonium batch experiments using the minimum number of experiments needed to produce defensible plutonium sorption values. The effects of organics on actinide sorption will be assessed, and available ground water data will be evaluated to address potential deficiencies in the sorption data base. Sorption data will be collected for minerals and ground-water chemistries resulting from planned thermal loads

3.2.8 Study 8.3.1.3.4.2 - Biological Sorption and Transport

The objectives of this study are to determine what effects microorganisms have on the movement of radioactive wastes (i.e., effects on transport) and to determine if microbial activities play a role significant enough to be included in a performance calculation for Yucca Mountain. The objectives will be accomplished by combining the results of laboratory experiments investigating the different mechanisms of microbial effects on transport (chelation, sorption, and colloidal interactions) with microbial analysis of samples collected from the ESF that will determine the numbers, metabolic activity, identity, and diversity of the indigenous population in Yucca Mountain.

A paper (Hersman, in prep [a]) is being developed that describes the biological sorption and transport field and laboratory studies, presents an overview of the current literature, and summarizes the results of the work discussed in Kieft et al. (in prep.) and Ringelberg et al. (in prep.)

A report (Kieft et al., in prep.) discusses the distribution of heterotrophic microorganisms within Yucca Mountain. Generally, with increasing distance in the ESF, the population of microorganisms decreases, and a greater percentage of gram negative microorganisms is present in the samples. Ringelberg et al. (in prep.) describes the phospholipid fatty-acid analysis performed on the ESF samples. The major findings indicate that sulfate reducers are present in

Yucca Mountain, wł. ıgnificant because of the role that these microorganisms play in the corrosion of primaı y c rs.

Chelated transport of iron through columns of unsaturated tuff is described in Story et al. (in prep.). The major finding of this report is that chelation significantly affects the transport of iron in crushed tuff. Therefore, transport of radioactive wastes may be enhanced by chelation.

Forecast: Planned work will support development of process models for site-scale unsaturated zone transport and waste package degradation that will be used in the total system performance assessment for the viability assessment. The work also supports the process model for site-scale saturated zone transport and parts of the waste containment and isolation strategy that concern limited corrosion and low humidity. The work identifies key actinide concentrations that may be reduced by depletion and dispersion.

3.2.9 Study 8.3.1.3.4.3 - Development of Sorption Models

The objectives of this study are to model the sorption experiments on rocks and minerals representing the proposed repository block and to derive a capability to predict sorption coefficients for key radionuclides under water-rock conditions not included within the experimental program.

Project geochemists summarized the data needs for sorption modeling in performance assessment. These data needs include the following:

- Isotherms for cation exchange between the alkali metals and alkaline earths present in UE-25 J#13 water and the minerals contained in Yucca Mountain Tuff.

- A predictive model for cation exchange based on mineralogy. A mineralogy based model of cation exchange will allow the use of the enormous and extensive mineralogy and petrology data base to arrive at a more accurate K_d as a function of stratigraphy.

- A hard-soft acid base model for bidentate attachment mechanisms. Bidentate attachment is possible for any multivalent metal ion. The consequence of a bidentate attachment mechanism is a heightened sensitivity to competition by metal ions that attach by a monovalent mechanism.

- Stoichiometries for the surface complexation of radionuclides on minerals and tuff. Measurements of the pH dependence and electrolyte concentration dependence of radionuclides are needed to develop a predictive surface complexation model.

- Measurements of the competition for surface complexation sites. Competitive surface complexation experiments are needed to fully understand the consequences of changes in ground-water composition on sorption

- Cause of the wet sieving effect. The observed reduction in sorption upon wet sieving could mean that the in situ sorption coefficient is significantly larger than laboratory measurements indicate. If the effect is caused by site saturation by trace components of ground water, ground-water analyses may have to be remeasured with specific attention to those components.

Forecast: This study will summarize the mechanistic models for sorption of radionuclides onto Yucca Mountain minerals. A summary report (describing the findings to date from Studies 8.3.1.3.4.1 and 8.3.1.3.4.3) will be prepared. For Study 8.3.1.3.4.3 the summary report will be on (a) summarizing available surface complexation models to describe radionuclide sorption, (b) evaluating the predictive capability of surface complexation models to describe radionuclide sorption onto pure minerals and whole rock, (c) assessing the value of spectroscopic studies for direct measurement of sorbed species and surfaces, and (d) describing the effects of organics on radionuclide sorption.

3.2.10 Study 8.3.1.3.5.1 - Dissolved Species Concentration Limits

The objective of this study is to determine the solubilities and speciation of important radioactive waste elements under conditions characteristic of the repository and along flow paths from the repository into the accessible environment.

Activity 8.3.1.3.5.1.1 - Solubility measurements. The objective of this activity is to specify the conditions under which solubility experiments will be conducted and then measure solubility limits of important waste elements under these conditions.

Project chemists continued to characterize neptunium and plutonium solid precipitates from solubility experiments from oversaturation. No evidence for a Np (IV) solid, either crystalline or amorphous, has been found. Furthermore, adding excess ferrous chloride to a neutral de-aerated solution of Np(V) in 3 mM sodium bicarbonate (to simulate UE-25 J#13 water conditions) showed no reduction to a Np(IV) solution/precipitation at room temperature. While a couple of solubility experiments from undersaturation have started using neptunium solids from oversaturation experiments, the main experiments using well characterized neptunium oxides (Np_2O_5 and NpO_2) solids in UE-25 J#13 and in sodium perchlorate ($NaClO_4$) electrolyte solutions are about to begin.

The importance of the oxidation state of the solubility-limiting solid (i.e., Np [V] versus Np [IV]) is that Np (IV) solids would have solubility 2 to 3 orders of magnitude less than NP (V) solids, so any evidence of formation of Np (IV) solids would benefit system performance.

Activity 8.3.1.3.5.1.2 - Speciation Measurements. The objective of this activity is to identify important aqueous species of waste elements under conditions described in Activity 8.3.1.3.5.1.1 and to determine their formation constants. Such thermodynamic information is essential to model the solubility of key radionuclides as described in Activity 8.3.1.3.5.1.3.

This activity has been deferred because of a lack of funding.

Activity 8.3.1.3.5.1.3 - Solubility modeling. The objective of this activity is to develop the thermodynamic models with a consistent data base needed to calculate waste element solubilities over the range of conditions expected at the site.

This activity has been deferred because of a lack of funding.

Forecast: Work will determine whether Np(IV) species may form under natural conditions at Yucca Mountain. Forcing conditions, including high temperature and control of oxidation-reduction potential, will be imposed and are intended to constrain both the kinetics and thermodynamics to test whether natural conditions at Yucca Mountain might be expected to produce Np(IV). Completion of these experiments should identify the final constraints on neptunium solubility.

3.2.11 Study 8.3.1.3.5.2 - Colloid Behavior

The objective of this study is to determine the stability of waste element colloids under expected site-specific conditions that might be encountered at the repository or along flow paths toward the accessible environment.

Activity 8.3.1.3.5.2.1 - Colloid formation characterization and stability. The objective of this activity is to determine formation and stability of waste element colloids.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.3.5.2.2 - Colloid modeling. The objective of this activity is to develop models and parameters to calculate natural colloid concentrations and stability and to describe the disposition of the waste element species as the colloids break up.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Further work on this study has been deferred because of funding constraints.

3.2.12 Study 8.3.1.3.6.1 - Dynamic Transport Column Experiments

The objective of this study is to measure the breakthrough or elution curve for tracers through tuff columns.

Activity 8.3.1.3.6.1.1 - Crushed tuff column experiments. The objective of this activity is to measure the rate of movement through crushed tuff columns of radionuclides relative to tritiated water and other well-defined chemical species or colloids.

No progress was made during the reporting period, this was an unfunded activity.

Activity 8.3.1.3.6.1.2 - Mass transfer kinetics. The objective of this activity is to determine the elution rate of radionuclides as a function of water velocity for crushed tuff columns (homogeneous system), solid rock columns (heterogeneous system), and for pure mineral samples.

No progress was made during the reporting period; this was an unfunded activity

Activity 8.3.1.3.6.1.3 - Unsaturated tuff columns. The objective of this activity is to measure the relative migration rate of radionuclides through partially unsaturated rock columns.

Neptunium experiments (to expand th data base) under varying degrees of saturation, were initiated using two tuff types (zeolitic and devitrified) and two bicarbonate waters. The purpose of the experiments was to expand the neptunium data base.

Activity 8.3.1.3.6.1.4 - Fractured tuff column studies. The objectives of this activity are to measure the transport and diffusion of radionuclides through naturally fractured tuff; and to examine the movement of tracers through naturally fractured Yucca Mountain cores to test the transport models.

Experiments were initiated to expand the data base for the fractured rock column experiments to increase confidence in findings on radionuclide retardation in fractures. Project scientists used autoradiography to assess the fracture-matrix coupling for alpha emitting radionuclides flowing through fractures.

Activity 8.3.1.3.6.1.5 Filtration. The objective of this activity is to quantify the filtration of colloids and particulates by the tuff as a function of particle or pore size using solid tuff cores and fractured cores.

A paper (Degueldre et al., in prep.) is being prepared that presents ground water colloid results from various geologic formations ranging from crystalline to sedimentary, saturated to unsaturated, organic rich to organic depleted. Colloid presence and mobility are explained on the basis of stability properties in the ground waters studied. Colloid concentration is a function of pH, oxidation reduction potential, cation and organic carbon concentrations, and the status of the chemical and physical steady state of the hydrogeochemical system.

Forecast: The results of transport experiments consisting of eluting radionuclides through saturated and unsaturated, crushed, fractured, and solid tuff columns will be summarized. This summary will assess the validity of using batch sorption distribution coefficients to describe the migration of radionuclides through fractured and solid tuff under various degrees of saturation. This summary will also assess the results of eluting colloids through tuff columns and the implications of those results for performance assessment calculations of radionuclide transport.

3.2.13 Study 8.3.1.3.6.2 - Diffusion

The objectives of this study are to measure the diffusivity and kinetics of adsorption in a purely diffusive system (i.e., no advection) from the uptake of radionuclides on intact tuff as a function of time; and to conduct scaling studies to determine the length to which the matrix diffusion model can be applied with confidence

Activity 8.3.1.3.6.2.1 - Uptake of radionuclides on rock beakers in a saturated system. The objective of this activity is to measure the uptake of radionuclides by rock beakers as a function of time.

No progress was made during the reporting period, this was an unfunded activity.

Activity 8.3.1.3.6.2.2 Diffusion through a saturated tuff slab. The objective of this activity is to measure the diffusion of radionuclides in a purely diffusive system (no advection) by observing the migration of radionuclides through slabs of Topopah Spring Tuff and Calico Hills zeolitic tuff

Five diffusion experiments began In each experiment, a diffusion cell was set up in which radionuclides are in contact with a coated fracture in one of the two chambers of the cell. In the other cell chamber (which is in contact with the tuff matrix) arrival of the radionuclides (which have diffused through the fracture and the matrix) was monitored. Evaluation and interpretation of results are in progress.

Activity 8.3.1.3.6.2.3 - Diffusion in an unsaturated tuff block The objectives of this activity are to determine the distribution of radioactivity in the unsaturated tuff matrix, using an unsaturated block of the Topopah Spring Tuff or the Calico Hills Formation, and to fit the uptake of radionuclides as a function of time to a diffusion model with reactions (sorption) to determine the diffusivities and rate constants

Experiments were started to measure diffusion coefficients as a function of saturation in zeolitic and devitrified tuffs for sorbing and nonsorbing radionuclides

Forecast: A report will be prepared that summarizes the results of the diffusion experiments and provides diffusion coefficients as a function of hydrologic unit, tuff moisture content, and temperature The report will describe the use of diffusion data to develop diffusion coefficients for radionuclides migrating through fractures into the tuff matrix in the presence of fracture coatings

3.2.14 Study 8.3.1.3.7.1 - Retardation Sensitivity Analysis

The objectives of this study are to develop a conceptual geochemical-geophysical description of Yucca Mountain based on the results, data, and information generated from the geochemistry, mineralogy-petrology, hydrology, and other pertinent Project tasks, and to determine which data are most important to make the cumulative, integrated transport calculations needed to meet the U.S. Nuclear Regulatory Commission (NRC) and U.S. Environmental Protection Agency regulations. Another important function of this study is to provide process models from which performance assessment abstractions and simplifications can be made.

The study is the focal synthesis activity for hydrology, mineralogy/petrology, sorption, dynamic transport, and solubility/speciation tasks by incorporating information from all these sources into abstracted site-scale transport models for both the unsaturated and saturated zones.

Activity 8.3.1.3.7.1.1 - Analysis of physical/chemical processes affecting transport. The objectives of this activity are to analyze processes that may affect transport, geochemical, physical, particulate, heat-load effects, and coupled phenomena; to support and develop those laboratory experiments needed to examine the physical and geochemical processes affecting radionuclide transport and other experimental activities under this program and the ESF tests; and to correlate and validate results obtained from laboratory, ESF, and field experimental results with transport calculations.

Development of the saturated-zone model continued to mature. Several different representations of stratigraphy, water-table, and boundary conditions were created in the Stratamodel geologic modeling packaged and converted into numerical grids using the GEOMESH grid-generation software. The main objective of this work was to include the most reliable data on water levels based on well observations, as well as the most up-to-date geologic interpretations. Grid resolution was kept relatively coarse so that a parameter estimation routine (PEST) could be used efficiently for the calibration. The PEST software requires numerous simulations to obtain the fit to the data, and thus computational efficiency is important.

Comparisons were begun of structured and unstructured grids as the result of an update meeting on Performance Assessment held the week of February 3, 1994 in Denver.

Following the meeting, a backup model was generated based on structured (finite difference) grids representing the same volume as the unstructured mesh (finite element). In addition, a resolution study was underway to determine the grid resolution necessary for saturated-zone studies. Using constant head-boundary conditions, the change in flux at the south boundary (and near a regulatory compliance boundary) was monitored as grid resolution was increased. Analysts determined that fluxes vary as much as 50 percent from the values computed at the coarsest resolution. This variation indicates that relatively fine grids will be required for the final calculations. Therefore, the approach to model calibration will be to perform a coarse-grid calibration to determine an approximate permeability distribution that will be used as the initial state for the final fine-grid calibrations.

Activity 8.3.1.3.7.1.2 - Geochemical/geophysical model of Yucca Mountain and integrated geochemical transport calculations. The objective of this activity is to perform calculations of radionuclide transport from the repository to the accessible environment using, as a basis, an integrated, conceptual geochemical geophysical model of Yucca Mountain.

The transport of neptunium-237, one of the key radionuclides in the inventory, was simulated using a comprehensive geochemical transport model (Viswanathan et al., 1977). The laboratory-scale speciation and sorption data for neptunium-237 were replicated using the reactive transport module of FEHM. The model correctly simulated the transition from sorbing neptunyl cation to non-sorbing carbonate complexes at higher pH values in typical Yucca Mountain fluids.

Furthermore, the effect of carbonate concentration in the fluid (varied by performing experiments in both UE-25 J#13 and UE-25 p#1 water) on speciation is also correctly represented in the model. An ion exchange sorption model was developed to capture the competitive sorption effects of major cations such as sodium, magnesium, and calcium. When speciation calculations are performed during simulations of transport through the unsaturated zone, the calculations directly incorporate the effects of geochemistry on transport of neptunium-237. Significant retardation of radionuclides occurred in the zeolitized CHn units as long as the unsaturated zone fluid had a carbonate concentration that resembled that of UE-25 J#13 fluid. Recent bicarbonate ion measurements for unsaturated-zone pore fluids reported by Yang et al. (1996) confirm this observation. Project hydrologists have corroborated retardation of neptunium-237 in the zeolitized CHn units using this reactive transport model, thereby reducing an uncertainty for one of the crucial radionuclides in the inventory.

Project scientists also investigated the influence of repository heat on the transport of neptunium-237. Together with the geochemical transport model discussed above, scientists examined whether the period during which thermohydrologic effects are likely to be most vigorous (the first 5000 years after waste emplacement) changed the transport predictions for neptunium-237 significantly from predictions that assumed isothermal conditions. Results indicate that since most of the neptunium-237 inventory is released from the repository in a period that exceeds the most important thermohydrologic effects, the influence of waste heat on radionuclide movement is minimal. This conclusion simplifies the execution of radionuclide transport predictions since it suggests that the predictions can be made assuming that isothermal conditions prevail.

The main caveat to this conclusion is that the influence of possible permanent changes to the hydrologic and transport properties needs to be studied. An activity to examine changes to the permeability and porosity distribution due to rock-water interactions is planned.

Activity 8.3.1.3.7.1.3 - Transport models and related support. The objectives of this activity are to verify the computer codes and to validate the models used in this study and to identify important contributors to the uncertainties in retardation calculations (sensitivity analyses).

Several code modifications were made to FEHM to accommodate needs of saturated-zone users. The first was to incorporate hydraulic head input and output instead of pressure. This will be very useful for linking this site-scale model to the USGS regional-scale model. The second modification was to implement an option for general time-varying boundary conditions. A sophisticated approach was necessary because multiple time-step adjustments were required to get cyclic fluxes in different parts of a model domain. A Bousinesq approximation was also incorporated as an option for saturated zone flow simulations. Lastly an output module for the PEST parameter estimation code was added.

Improvements to the reactive-transport model in FEHM increased the efficiency of simulation of the transport of multiple, interacting species in the solid, aqueous, or gas phases (Viswanathan et al., 1997). Equilibrium and kinetic reactions were recast as systems of equations that can be solved as algebraic equations at each node to improve computational efficiency. This code development effort allowed the simulation of the geochemical transport behavior of neptunium-237 (see previous discussion under Activity 8.3.1.3.7.1.2).

Forecast: The importance of uncertainty in conceptual models and parameters for flow and transport in the unsaturated and saturated zones will be tested in support of performance assessment. Information from hydrology, mineralogy-petrology, sorption, dynamic transport, and solubility-speciation studies will be incorporated into abstracted site-scale transport models for both the unsaturated and saturated zones. Model studies will be performed to create abstractions suitable for performance assessment, including testing the use of simple sorption and solubility models against complex, reactive transport models. Reactive transport models for neptunium, uranium, and technetium will be developed for the unsaturated zone. For technetium in the saturated zone, a model will be developed to test whether simple solubility/speciation and sorption models can be used in model abstraction. The reactive transport models will include solubility-speciation, dissolution-precipitation, and reduction-oxidation reactions in the presence of repository waste heat and the introduction of manmade materials.

Process models for radionuclide transport will be revised by incorporating and summarizing progress through FY 1996 and the first half of FY 1997. This work will include updating documentation of transport codes, meshes, etc., as well as developing intellectual insights into the processes involved in the migration of radionuclides through the natural barrier. Processes that affect the transport of radionuclides at Yucca Mountain will be studied to support the development of simplifying assumptions made by performance assessment for transport models. Retardation processes and specific data accuracy requirements will be identified; these specifications will help assess whether site characterization data-gathering activities performed to date have been adequate.

Information will be incorporated from hydrology, mineralogy-petrology, sorption, dynamic transport, and solubility-specification tasks into site-scale transport models for the unsaturated zone. Where data are unavailable or insufficient to accurately set parameter values, sensitivity analyses will be performed to determine the effects that variations in parameters or processes have on transport from the potential repository. Numerical flow and transport models developed for performance assessment will be used, along with enhancements, including reactive chemistry, that have been developed in previous years in this task. Predictions of radionuclide migration will be produced, along with uncertainties that will point to further data needs in the sorption, dynamic transport, hydrology, and geological framework tasks.

Two reports will be completed. One will include a conceptual model of the transport of radionuclides in the unsaturated zone at Yucca Mountain, including two- and three-dimensional integrated transport calculations and sensitivity analyses. Both equivalent-continuum and dual-permeability calculations will be made, incorporating discrete structural features as available and appropriate. The model will incorporate and consider isotopic data indicative of processes and rates of fluid flow or residence times in the mountain. The model will update three-dimensional transport simulations for radionuclides including, but not limited to, neptunium, plutonium, and technetium under ambient and thermally perturbed scenarios for two future climates and accompanying infiltration fluxes. The report will also be coupled to appropriate near-field source term and thermal model(s). Where necessary, reactive transport models for key radionuclides such as neptunium and uranium will be developed to better capture the solubility-speciation and sorption data available for these radionuclides.

The second report will summarize development and implementation of the model for the reactive transport capabilities of the FEHM computer code. New options will be summarized, including speciation, complexation, and sorption reactive transport capabilities. Updated user documentation will also be provided, along with a series of example problems.

3.2.15 Study 8.3.1.3.7.2 - Demonstration of Applicability of Laboratory Data to Repository Transport Calculations

The objective of this study is to outline the strategy that will be used to demonstrate the validity of the laboratory generated geochemical data and the transport calculations using that data.

Activity 8.3.1.3.7.2.1 - Intermediate-scale experiments. The objective of this activity is to conduct experiments at a scale larger than a laboratory but with sufficient control on material and boundary conditions to test how increased spatial scale affects water flow and radionuclide transport in unsaturated porous media.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.3.7.2.2 - Field-scale experiments to study radionuclide transport at Yucca Mountain. The objective of this activity is to evaluate the validity of laboratory-derived data and models for radionuclide transport at the Yucca Mountain site by conducting tests in the bedded tuffs of the Calico Hills unit underlying the Topopah Spring unit.

Two sites have been identified as possible locations for an unsaturated zone transport test: P-tunnel and Busted Butte. P-tunnel scoping calculations demonstrate the feasibility and benefits of an unsaturated zone transport test to the project partially completed prior to license application. Field reconnaissance of the northern and southern Busted Butte areas conducted in March 1997 identified the southern section as a favorable site for unsaturated zone testing in the Calico Hills Formation.

A summary of the results from the first C-wells reactive tracer tests completed in early 1997 was submitted to the Spring 1997 American Geophysical Union meeting (Turin and Reimus, in prep.). The objective of the test was to provide a field validation of reactive tracer transport in saturated, fractured tuffs at Yucca Mountain immediately east-southeast of the footprint of the potential repository site. Detailed description of the results can be found in Milestone No. SP2370M4 (March 14, 1997). A forced-gradient field tracer test involving simultaneous injection of nonsorbing solute tracers of different diffusivities (pentafluorobenzoic acid and bromide), a sorbing solute tracer (lithium ion), and a colloidal tracer (polystyrene microspheres) was conducted at the C-Hole complex in saturated, fractured tuff in the vicinity of Yucca Mountain, Nevada. The test objectives were to (1) test and validate a conceptual model for contaminant transport in fractured media, and (2) assess the validity of using laboratory-derived sorption parameters to predict reactive tracer transport at the field scale in unsaturated tuffs. All tracer breakthrough curves were bimodal in nature, indicating multiple flow pathways. Matrix diffusion was apparent in each pathway based on differences in the breakthrough curves of the two nonsorbing solutes. Sorption of lithium ion was deduced by comparing its response to the response of the nonsorbing tracers. Parameters for dispersion, matrix diffusion, and sorption in each major pathway were estimated by simultaneously fitting the data for all solute tracers using a semi-analytical model. The recovery of microspheres was significantly lower than that of the solutes, but a small fraction of them arrived earlier than the solutes. The sorption parameters obtained for the lithium ion were in good agreement with laboratory measurements.

Activity 8.3.1.3.7.2.3 - Natural analog studies of radionuclide transport. The objective of this activity is to use natural analog studies and data generated by natural analog studies to support long-term calculations of radionuclide transport using laboratory data and radionuclide transport models.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.3.7.2.4 - Data on radionuclide transport from other U.S. Department of Energy sites (Anthropogenic Analogs). The objective of this activity is to evaluate the validity of laboratory-derived data and models for radionuclide transport at the Yucca Mountain site by obtaining data collected at other United States DOE sites on radionuclide distribution in geologic systems.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: For the unsaturated zone, a test plan was being prepared and will be submitted for review (SP343TM4; 6 Jun 97); the testing program will support the development, calibration, and validation of the unsaturated zone transport process model to be asbstracted for total system performance assessment and for the license application. The southern Busted Butte area, south of the potential Yucca Mountain repository, has been recommended as a favorable site where rocks from the Calico Hills Formation are exposed.

For the saturated zone, the Project is considering several activities to assess the ability to predict sorbing tracer transport at the field scale using laboratory derived sorption parameters in conjunction with conceptual transport models for FY 1998. The effectiveness of matrix diffusion as a solute retardation mechanism and the potential for colloid transport in the saturated zone will also be evaluated. Continued testing at the C-Hole complex supports evaluation of key attributes of the draft DOE Waste Containment and Isolation Strategy that is concerned with dilution, dispersion and retardation in the saturated zone as well as development of the saturated zone flow and transport model needed to support total system performance assessment for the viability assessment and total system performance assessment for the license application.

3.2.16 Study 8.3.1.3.8.1 - Gaseous Radionuclide Transport Calculations and Measurements

The objectives of this study are to calculate the rates of transport of gaseous radionuclide species between the repository and the accessible environment considering the various driving forces and retardation mechanisms that may exist; and to experimentally verify potential existing models of gaseous radionuclide transport and retardation that are used to assess radionuclide release to the environment.

No study plan will be developed; the scope of this study is encompassed by Study 8.3.1.2.2.6 and Activity 8.3.1.3.7.1.3; see Sections 3.1.10 and 3.2.14 of this progress report.

Forecast: Further work on this study has been deferred because of funding constraints.

3.2.17 Related International Geochemical Work

No progress occurred during the reporting period. As of November 8, 1995, the subsidiary agreements under which the cooperative work had been conducted were terminated and all international collaboration was discontinued.

The OCRWM had bilateral agreements with Canada (Atomic Energy of Canada Limited [AECL]), Switzerland (Swiss National Cooperative for the Storage of Radioactive Waste [Nagra]), and Sweden (Swedish Nuclear Fuel and Waste Management Company [SKB]) and has participated in activities of international organizations such as the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA), European Commission (EC), and the International Atomic Energy Agency (IAEA).

3.3 ROCK CHARACTERISTICS (SCP SECTION 8.3.1.4)

The changes to the Rock Characteristics Program since the SCP was issued are summarized in Appendix A, Section A.1.3.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpln, Tpcpv3) are found in Buesch et al. (1996a).

3.3.1 Activity 8.3.1.4.1.1 - Development of an Integrated Drilling Program

The objectives of this activity are to (a) ensure representativeness of data acquired during surface-based site characterization activities and that data represent the range of phenomena and structural characteristics needed for performance assessment, and (b) integrate and prioritize surface-based activities to produce a schedule that best addresses representativeness and efficacy concerns, given resource constraints.

There will be no study plan developed for this activity.

A Daily Field Activity Report has been distributed throughout YMSCO and to NRC.

Field Work Packages are being developed for the following activities:

- C-Hole Complex Period 3 Tracer Testing
- Installation of Seismic Gauges at USW UZ-14, USW SD-7 and Large Block Test Site
- Installation of SEAMIST™ System at USW UZ-14
- Crest Borehole USW SD-6
- Surface Based Borehole Security at YMP
- Surface Based Borehole Monitoring & Testing
- Seismic Monitoring for YMP
- Borehole UE-25 WT#17 & UE-25 WT#3 Cleaning, Rehabilitation and Testing
- Borehole USW WT-24 Drilling and Testing.

Forecast: Drilling activities for FY 1997 and FY 1998 have been prioritized on the basis of the needs of the hydrology, geology and repository program. Distribution of planning documents for the following activities is expected during FY 1997:

- C-Hole Work Period 3-Tracer Testing
- Borehole Workover and Pump Testing at UE-25 WT#17 & USW WT#3
- FY 1997 Master Plan for Trench and Test Pit Reclamation
- Flexible borehole liner installation at USW UZ-14
- Crest Borehole USW SD-6
- 2nd Crest Borehole (not yet named)
- USW WT-24 Drilling & Testing

3.3.2 Activity 8.3.1.4.1.2 - Integration of Geophysical Activities

The objective of this activity is to provide a mechanism for information exchange, an analysis of data and other technical information, and an overview of planned geophysical site characterization activities.

Tasks in geophysics focused on evaluating existing geophysical data through processing and modeling. Most of the emphasis was placed on the aeromagnetic data and regional seismic lines. The evaluation of the magnetic data showed that the survey was not of sufficient quality to allow the determination of the depth to basement. Most of the information contained in the data came from shallow structures and from the lithology above the basement. After review of the seismic reflection data, an independent evaluation concluded that further processing of the data would not significantly improve the results. Modeling is in progress to assess the confidence in the current interpretations.

Records packages were organized, cataloged, reviewed, authenticated, and submitted for the following Yucca Mountain boreholes:

USW NRG-7/7A
USW SD-7
USW SD-9
USW SD-12
USW ONC#1
NRG boreholes (supplement)

Records packages for the three FY 1996 reports were organized, cataloged, reviewed, authenticated, and submitted (CRWMS M&O, 1996g, 1996h, and 1996i).

Records packages for the following Yucca Mountain boreholes remain to be completed:

UE-25 c#3
UE-25 UZ#16

Forecast: No new surface geophysical data collection is planned in FY 1997. The near-term focus will be on cataloging and storing all acquired borehole geophysical logging data for future use. Efforts will continue to integrate geophysical data with geological information and interpretations to develop an understanding of the geological framework that is consistent with the total data set. The modeling discussed in the preceding paragraph will be completed and documented in April 1997. The results will be discussed in Progress Report #17.

3.3.3 Study 8.3.1.4.2.1 - Characterization of the Vertical and Lateral Distribution of Stratigraphic Units Within the Site Area

The objective of this study is to determine the vertical and lateral variability and emplacement history of stratigraphic units and lithostratigraphic subunits within the Yucca Mountain site area.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpln, Tptpv3) are found in Buesch et al. (1996a).

Activity 8.3.1.4.2.1.1 - Surface and subsurface stratigraphic studies of the host rock and surrounding units. The objective of this activity is to determine the spatial distribution, history, and characteristics of stratigraphic units within the Paintbrush Group, Calico Hills Formation, Crater Flat Group, and possibly older volcanic rocks within the site area.

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One of the primary purposes of this activity is to understand the lateral variations in hydrogeologic properties of various units within the Tertiary volcanic sequence. Recent techniques applied to correlate lithologic features with hydrogeologic properties, coupled with newly calculated porosity and saturation logs, have provided a means for determining the vertical and lateral continuity of potentially significant hydrogeologic layers (some of which are locally less than 1 m thick) and for correlating units across a site where only selected boreholes have been cored. Linkage of lithologic features, hydrogeologic matrix properties, and geophysical logs helps to constrain variations in matrix and bulk-rock porosity, pore-space geometry, fract e-induced porosity, and water occurrence in pore space or as structurally bound water. All the features influence the flow of moisture through the site. Some recent results of these studies were presented in Buesch et al (1996b)

Several key findings of the studies involving correlation of lithostratigraphic and geophysical logs were described in a memorandum report submitted to YMSCO (USGS, 1997b). These findings are summarized here:

1. Hydrogeologic and thermal-mechanical unit models traditionally have relied on the spatial position of moderately to densely welded intervals to define thicknesses of altered and unaltered zones. Several formations, particularly the Tiva Canyon Tuff, Topopah Spring Tuff, and Prow Pass Tuff, have been the focus of matrix hydrogeologic properties studies, and have significantly thick sections of partially to densely welded rocks that have experienced vapor-phase alteration or corrosion and mineralization. [Note: Vapor-phase alteration or corrosion are terms intended to refer to dissolution of pumice fragments by hot vapors before moderately welded tuffs cool. Recently, the word corrosion has been used to distinguish this micro-scale process from the larger-scale formation of lithophysal cavities.] These intervals are minimally altered and characteristically have physical properties that more closely resemble crystallized and welded rock sections

2. Contacts between vitric and crystallized boundaries (including zones of vapor-phase alteration and mineralization) in several formations commonly are marked by intervals several millimeters to several meters thick that contain clay alteration and may have hydrologic significance

3. The contact between the Calico Hills Formation and the Prow Pass Tuff in borehole USW G-2 is an erosional-depositional contact and not a fault contact as previously interpreted. Therefore, a total thickness for the Calico Hills Formation can be derived from this borehole. This finding requires at least some modification of the geometric relation of faults beneath and parallel to Sever Wash

4. The entire thickness of the Topopah Spring Tuff occurs in borehole UE-25 p#1. On the basis of this interpretation, no significant fault (currently shown with about 200 m of offset) is required within the uppermost part of UE-25 p#1 to juxtapose the Tiva Canyon Tuff and Topopah Spring Tuff

5. A distinctive and abrupt decrease in density and corresponding increase in calculated porosity at the Topopah Spring Tuff crystal-poor lower nonlithophysal zone-vitric/densely welded interval contact (Tptpln-pv3) occurs commonly throughout Yucca Mountain and can be used to identify as much as several meters of intensely argillic and zeolitic rock near the top of the Tptpv3. This feature has been noted as a petrophysical zone on geophysical logs, but the occurrence of clays and zeolites at or near the Tptpln-pv3 contact has been documented previously in only a few boreholes. This interval of alteration commonly is coincident with "wash outs" in many boreholes and, therefore, its mineralogic control and lateral extent has not been previously mapped in the subsurface. In several boreholes, this alteration occurs extensively along fractures in the vitric densely welded rocks (Tptpv3). An important implication of this occurrence is that lower densities of the alteration minerals would tend to produce anonymously higher porosities, if calculated porosity models do not account for these minerals.

6. The contact between the densely welded vitric rocks near the base of the Topopah Spring Tuff and pervasively zeolitic rocks of the Calico Hills nonwelded hydrogeologic unit (CHn) (referred to as the vitric-zeolitic boundary) has been interpreted in several boreholes to occur coincident with the Tptpln-pv3 contact, based on the abundance of zeolite-filled fractures in the Tptpv3. Therefore, this contact is not exclusively confined to the CHn hydrogeologic unit as previously thought.

7. Boreholes UE-25 WT#14 and UE-25 p#1 in Midway Valley indicate that the entire section of rocks from the Tiva Canyon Tuff down to the pre-Pah Canyon bedded tuff has been eroded.

8. Several contacts indicate that bedded tuff overlies crystallized rocks of the subjacent formation. This relation indicates that the upper vitric, nonwelded to partially welded tuff was eroded before deposition of the overlying bedded tuffs. This conclusion is predicated on the assumption that rocks did not weld and crystallize to the very top of formations.

Lithostratigraphic descriptions have been provided to support several mapping activities in the southern part of the main drift and south ramp of the ESF (see Section 3.3.4 of this progress report, Activity 8.3.1.4.2.2.4). Many of these same rocks have been traversed in the north ramp and along the main drift. In support of matrix-hydrologic properties testing (Section 3.1.7 of this progress report, Activity 8.3.1.2.2.3.1), staff geologists have provided possible lithostratigraphic and structural context for the relatively systematic increases and decreases in porosity that have been documented in rocks from the middle nonlithophysal zone of the Topopah Spring Tuff (repository horizon) exposed in the main drift of the ESF. Variations in porosity might result from (a) small amounts of trapped, intergranular gas (vapor) associated with the occurrence of the lithophysae-bearing subzone of the middle nonlithophysal, even though there are no well developed macroscopic lithophysae in the main drift, (b) small amounts of trapped gas (vapor) that result from slightly different cooling regimes associated with the formation of the close-spaced high-angle fractures, or (c) microscopic fracturing associated with possible tectonic adjustments of the close-spaced, high-angle fractures.

Project geologists examined three vertical boreholes in the crown of the northern and southern Ghost Dance Fault alcoves and confirmed that the contact of the upper lithophysal (Tptpul) and middle nonlithophysal (Tptpmn) zones of the Topopah Spring Tuff is within 1 m of the contact predicted by the three-dimensional lithostratigraphic model YMP.R2.0 (Buesch et al., 1996b). Staff geologists also examined a horizontal borehole in the Northern Ghost Dance Fault Alcove to establish the location of the Ghost Dance fault. The length of this borehole was 60 m. The eastern end of the borehole extension penetrated to about 165 m east of the ESF main drift. The video log and core from the borehole indicated the main trace of the Ghost Dance fault was located approximately 154 m east of the ESF main drift; the extended boring penetrated approximately 11 m into the footwall of the fault. The hanging wall (west side) of the fault consists of an 11-m wide zone of variably broken rocks with five breccia zones and the foot wall (east side) consists of 1 m of broken rock adjacent to the main trace. This geometry of distributed fracturing is consistent with observations at the land surface. Also, projection from the surface to the alcove intercept indicates that the fault is very nearly vertical. This work will support ESF design and the air-permeability and hydrochemistry tests in the Northern and Southern Ghost Dance Fault Alcoves, respectively (Section 3.1.8 of this progress report).

Activity 8.3.1.4.2.1.2 - Surface-based geophysical surveys. The objective of this activity is to improve confidence in stratigraphic models of Yucca Mountain by incorporating geophysical constraints.

Evaluation of the high resolution seismic data (Majer et al., 1996) in the repository region identified no seismic reflections from the interface between Paleozoic and Tertiary units. The lack of reflections is still attributed to the combination of the small amount of energy penetrating to depth (high attenuation of the tuffs), and possibly a smaller contrast in the acoustic impedances between the Paleozoic rocks and the overlying tuffs.

The amount of offset of the Paleozoic-Cenozoic unconformity remains indeterminate, because geologic control is lacking (DOE, 1997e) and the energy sources for the shallow seismic work were not strong enough to unequivocally resolve such a deep feature. Three-dimensional gravity models are consistent with minimal offset of the unconformity beneath the repository block (comparable to surface offsets of a few meters), but data from regional seismic reflection studies (DOE, 1997e) were previously interpreted as a reflector corresponding to an offset of the unconformity of about 1500 m by the Ghost Dance fault (Brocher et al., 1996). In a meeting of the scientists on December 16, 1996, at the Lawrence Berkeley National Laboratories, it was agreed that the possible escarpment interpreted in the Paleozoic unconformity under Yucca Mountain could not possibly be the Ghost Dance fault. The orientation and geometry of this feature are not compatible with that of the Ghost Dance fault.

Activity 8.3.1.4.2.1.3 - Borehole geophysical surveys. The objectives of this activity are to help define and refine the location and character of lithostratigraphic units and contacts between units and to determine the distribution of rock properties within lithostratigraphic units.

Data in the borehole-fracture data base that was collected between 1979 and 1985 by inspecting borehole walls using borehole television cameras and later technically reviewed in 1995, was reviewed for compliance with the technical review, reformatted to a standard.

organized, and submitted to the Technical Data Base. The borehole data base includes fracture data from 23 boreholes at Yucca Mountain. Boreholes include USW G-2, USW G-4, USW H-3, USW H-4, USW H-6, USW UZ-1, USW WT-1, USW WT-10, USW WT-11, USW WT-2, USW WT-7, UE-25 C#1, UE-25 C#2, UE-25 C#3, UE-25 UZ#4, UE-25 UZ#5, UE-25 WT#14, UE-25 WT#15, UE-25 #16, UE-25 #17, UE-25 WT#18, UE-25 WT#4, UE-25 WT#6.

A comprehensive data transcription verification was completed for the digital data base of geophysical logs collected from boreholes at Yucca Mountain prior to 1991. The transcription verification relied heavily on the assessment performed in August 1995 by the Borehole Geophysical Data Technical Assessment Committee (see Activity 8.3.1.4.2.1.3 in Section 3.3.3 of Progress Report #15). Two methods were used to perform the transcription verification. The first method consisted of six steps: (1) locate the original field copies of the borehole geophysical logs; (2) sort logs based on type, time, and run; (3) identify those logs used in the data base; (4) review the pattern of each trace throughout its length; (5) measure and compare selected point values from the image files, the original field prints, and the digital data; and (6) document the results of the transcription verification on review sheets for each borehole. The second method was applied only to logs that had a linear horizontal scale and involved use of a FORTRAN program and scale-manipulation tools of AutoCad to scale digital data from the data base back to the original scale of the field prints. Once properly scaled, the generated log traces were overlain onto the original field prints to make direct comparisons. Inconsistencies between the original field prints and the information in the data base were rare. Of about 400 log intervals of 41 boreholes checked, only 6 log intervals did not match the data base. These included two epithermal neutron porosity logs of UE-25 a#4, an electric log of UE-25 b#1, a caliper log of USW UZ-6, a dielectric log of UE-25 WT#17, and a gamma ray log of USW VH-1. However, only the log from UE-25 WT#17 had been used to calculate porosity and water content.

Activity 8.3.1.4.2.1.4 - Petrophysical properties testing. The objective of this activity is to provide geophysical and rock property data to be used in the interpretation of surface-based and borehole geophysical surveys.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.4.2.1.5 - Magnetic properties and stratigraphic correlations. The objective of this activity is to provide magnetic property data to aid in the interpretation of volcanic stratigraphy and structure of rock units, to use paleomagnetic directions to provide orientations for drill core segments, and to assess the rotation of rock units in relation to the geologic structures of Yucca Mountain from paleomagnetic indications.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Stratigraphic information will be provided for continued development of the three-dimensional, geologic-framework model. This activity will consist of continued correlation of lithostratigraphic features, borehole geophysics, and borehole video logs for the 32 lithostratigraphic contacts in 100 boreholes that are used in the geologic-framework model

3.3.4 Study 8.3.1.4.2.2 - Characterization of the Structural Features Within the Site Area

The objective of this study is to determine the frequency, distribution, characteristics, and relative chronology of structural features within the Yucca Mountain site area.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpll, Tptpmn) are found in Buesch et al. (1996a).

Activity 8.3.1.4.2.2.1 - Geologic mapping of zonal features in the Paintbrush Tuff. The objectives of this activity are to map zonal variations within exposed tuffs that will aid in the identification of structural displacement at a scale of 10 m or less; and to detect subtle changes in structural styles.

Field checking and revision of a preliminary site area map at 1:24,000-scale was initiated. The site area coincides with the area encompassed by the three-dimensional geologic framework computer model, version ISM2.0 (see Section 3.3.5 of this progress report). The study area extends from approximately the Prow of Yucca Mountain on the north to Busted Butte on the south, and from Windy Wash on the west to Fortymile Wash on the east. The new mapping effort concentrates on areas that may hold keys to both the structural and stratigraphic development of Yucca Mountain. Areas mapped in detail during the reporting period include Busted Butte, Fran Ridge, Alice Point, Dune Wash, UE-25 WT#12 basin, Iron Ridge, Crater Flat, upper Black Glass Canyon, upper Fortymile Wash, upper Yucca Wash, upper Teacup Wash, upper Fatigue Wash, Castellated Ridge, and the Prow.

The new mapping is delineating the structural complexities associated with both the hanging wall and footwall deformation associated with the major block-bounding faults, such as the Solitario Canyon, Fatigue Wash, Windy Wash, and Iron Ridge faults. These well known north-trending, block-bounding faults often are linked together by northwest-trending "relay" faults and associated structures that act to disperse the regional tectonic strain across several of the faults. Relay faults at Yucca Mountain are northwest-striking faults that intersect north-striking block-bounding faults and transfer displacement between two block-bounding faults. Relay faults provide a kinematic link between adjacent block-bounding faults. The block-bounding faults generally are west-dipping, down-to-the-west fault zones with numerous secondary faults that separate large blocks of east-dipping volcanic strata. Previous workers recognized that within the fault zones there are west-dipping structural panels. Because the trailing edges of the hanging walls of the north trending block-bounding faults are commonly intensely faulted and more easily eroded than adjacent rock, they are buried in the wash bottoms. However, the new mapping has better defined the footwall (eastern margin of the fault zones) deformation.

In general mapping associated with the FY 1996 central block map (Day et al., in press), as well as the current FY 1997 mapping campaign for the site map, has revealed that numerous intrablock faults, such as the Ghost Dance Fault, the Abandoned Wash Fault, and the Busted Butte Fault, are narrow zones at depth that "horsetail" toward the surface into a series of bifurcating faults. The new mapping has shown that this same pattern also exists for the block-bounding faults. The block-bounding fault zones are west-dipping, upward-widening zones with dips that vary from relatively shallow (45-50°) to relatively steep (75-85°). The total offset associated with the block bounding faults is a combination of down-to-the-west normal faulting and sinistral strike-slip motion, resulting in a net oblique-slip motion down to the southwest. In response to this oblique-slip motion, northwest-striking fault segments are the sites of releasing bends, commonly characterized by a gentler overall fault dip and a dramatic increase in the number of secondary normal faults preserved in the footwall. Northeast-striking fault segments are the sites of restraining bends, and are characterized by overall steeper, narrower fault zone that locally exhibit high-angle reverse faults.

The importance of this fundamental understanding of the fault geometry can be applied directly when trying to understand the amount of footwall deformation associated with the Solitario Canyon fault, which bounds the western margin of the repository area. The deformation within the footwall of the Solitario Canyon fault varies greatly, albeit systematically, along its trace. The footwall contains areas with relatively high and relatively moderate amounts of rock damage and associated fracturing along its trace. Further, fracturing of the footwall rock mass increases near the northwest-trending relay faults (south of the repository area). The bulk permeability should increase in the rock mass adjacent to and within these areas of high amounts of deformation. This higher bulk permeability might significantly affect the amount of percolation flux at the repository depths, as well as the coherency of the rock mass.

In addition to the more general question of the geometry of faults at Yucca Mountain the new bedrock geologic mapping has delineated the structural complexities associated with the Dune Wash graben. The Dune Wash graben is a northwest-trending structure bounded on the east side by the down-to-the west, block-bounding Dune Wash fault. The western margin of the graben is bounded by a down-to-the-east block-bounding fault zone with at least 122 m of down-to-the-east displacement, which is equal to or greater than that of the more widely known Dune Wash fault. The down-to-the-east fault zone is approximately 100 m wide and consists of tectonically brecciated and juxtaposed units of the Topopah Spring Tuff. Areas within the fault zone exhibit higher degrees of oxidation and alteration, as well as silicification, relative to other faults in the vicinity. This fault zone splays to the south into several faults with down-to-the-east displacement southwest of borehole UE-25 WT#17. The splays each have more than 33 m of displacement. Within the interior of the graben are numerous smaller horst-and-graben structural blocks with strata dipping dominantly to the east. The interior of the graben is highly complex with numerous, discontinuous, steeply dipping faults. The northern end of the Dune Wash graben dies out just south of Abandoned Wash (south of borehole USW WT-1). The deformation within the graben merges with that typical of hanging wall deformation associated with the Dune Wash fault. This style of hanging wall deformation associated with block-bounding faults is similar to that on the Bow Ridge and Solitario Canyon faults inasmuch as the dips of the strata increase, or roll over, into the block bounding faults. The southern end of the Dune Wash graben, which is buried beneath Quaternary deposits southwest of Busted Butte, seems to terminate against the down-to-the west Paintbrush Canyon fault.

As well as the structural insights gained, the new mapping has placed important constraints on the stratigraphic and regional tectonic history of Yucca Mountain. Mapping of areas containing the Rainier Mesa Tuff in Dune Wash and Solitario Canyon (Plug Hill) has revealed that the angular unconformity between the 12.7 million-year-old Tiva Canyon Tuff and the 11.45 million-year-old Rainier Mesa Tuff is relatively minor. There is no evidence to support a major angular unconformity at the top of the Tiva Canyon Tuff at Plug Hill. Flattening foliations within the Tiva Canyon Tuff dip at about 12 degrees to the east, which is similar in magnitude to the dip of the welding contact in the overlying Rainier Mesa. This is contrary to mapping by Scott and Bonk (1984) who depicted the basal contact of the Rainier Mesa Tuff as wrapping around and down cutting into the top of the Tiva Canyon Tuff, which implies a significant amount of erosion and post-Tiva Canyon Tuff, pre-Rainier Mesa Tuff regional deformation. The "down cutting" depicted by Scott and Bonk (1984) is simply a down-to-the-southwest fault that cuts the Tiva Canyon and Rainier Mesa tuffs, with the same amount of relative offset of units within each formation. Therefore, there is no significant amount of erosion at this contact.

There is a 5- to 8-degree structural unconformity at the base of the Rainier Mesa Tuff in the exposures at the south end of Dune Wash. There, an erosional unconformity between a bedded tuff and the top of the Tiva Canyon Tuff is exposed. These two units have the same dips, which implies there was no significant deformation between the deposition of these two units. Above the bedded tuff lies a nonwelded massive horizon of the Rainier Mesa Tuff, and there is a 5- to 8-degree structural discordance in their dips. Therefore, there was a modest amount of regional structural deformation that post-dated deposition of the Tiva Canyon Tuff but pre-dated that of the Rainier Mesa Tuff in the southern end of the Dune Wash graben.

Study of Fracturing Related to Faulting at Busted Butte

Collection of fault and fracture data (orientation, length, termination, aperture characteristics, etc.) has been initiated along the south side of Busted Butte as part of a detailed study of the interaction between faulting and fracturing associated with block-bounding faults, as well as to supplement ongoing site fracture studies. Fracture data have been collected in parts of the lower vitric section of the Tiva Canyon Tuff, the undifferentiated bedded tuffs underlying the Tiva Canyon Tuff, and the crystal-rich and upper lithophysal units of the Topopah Spring Tuff from the hanging wall of the major fault cutting Busted Butte. Sketch mapping and drawing of preliminary cross sections of the major fault have begun, with detailed mapping to commence following receipt of detailed orthophotographic base maps from the technical data base. Preliminary cross sections through Busted Butte were prepared, and a spreadsheet was formatted to store the fracture data. Data analysis began with various stereographic projections and rose diagrams of the fracture-orientation measurements and histogram analysis of other parameters.

Activity 8.3.1.4.2.2.2. Surface fracture network studies. The objective of this activity is to provide measurements and analyses of fracture networks to support modeling of potential hydrologic flowpaths, particularly in unsaturated zones. Applications are also expected to aid development of tectonic models and determination of the mechanical response of fractured rock to excavation and thermal loading.

A report has been prepared that integrates all quality-assured fracture data from surface studies, the ESF, and from boreholes (Sweetkind et al., 1997). These data have been compiled, integrated and interpreted to define fracture characteristics of each model layer within the site-scale, three-dimensional, unsaturated-zone flow model (see Section 3.1.13 of this progress report). Only a very limited amount of non-quality-assured data from boreholes (USW G-1, USW G-3/GU-3, and USW G-4) was used in this analysis. These borehole fracture data were selected for comparison with the quality-assured fracture data because (1) both television logs and core data are available for these boreholes, (2) these boreholes penetrate lithostratigraphic units equivalent to almost all of the model units in the site-scale unsaturated zone flow model, and (3) these data are readily available in published form. Preliminary comparison of the various data sets reveals general correspondence between various data collection methods and highlights the importance of lithostratigraphic controls, such as degree of welding, on fracture character. These integrated field data also are a critical part of the calculation of input parameters to three-dimensional fracture-network models, including number of fracture sets, mean orientation of each set, and dispersion about the mean; fracture trace-length distribution; a measure of fracture intensity; and a spatial model of fracture-network geometry (Sweetkind et al., in press).

In support of hydrologic models of the unsaturated zone at Yucca Mountain, new fracture data have been collected from the zeolitized rocks of the Calico Hills Formation from surface outcrops and borehole television logs. Surface fracture data consist of about 260 measured discontinuities from outcrops in the Prow Pass area to the northwest of Yucca Mountain. An additional 40 discontinuities were identified from borehole television logs from wells near the potential repository. Surface and subsurface data reveal the presence of two principal fracture sets, a prominent northwest-striking set and a slightly less well-developed northeast striking set. Perhaps the most important fracture characteristic within the zeolitized Calico Hills Formation is that joints appear to occur in zones. This characteristic was observed in the vicinity of Prow Pass where the northwest-striking joints occur in widely spaced northwest-trending zones. Each zone consists of a number of northwest-striking joints that are typically large (5 to 10 m long) and closely spaced (0.5 to 1 m). These northwest-trending zones are spaced from 50 to more than 100 m apart; in between the zones the spacing of the northwest-striking joints is 2 to 4 m or more. A zone of northeast-striking joints is present in borehole USW UZ-14 at a depth of 1477 to 1500 ft (450 to 457 m). These joints appear to be large (in many instances being continuous for 1.5 m or more within the hole), and have true spacing (corrected for the intercept angle between the borehole and the fracture set) of between 0.3 and 0.6 m. No evidence for a closely spaced zone of joints was found in either borehole USW SD-7 or USW SD-12.

Fracture data also are being used to evaluate the structural significance and characterize the possible hydrologic pathways used by water that has carried chlorine-36 from weapons testing to sample locations in the north ramp and main drift of the ESF (see Section 3.1.6 of this progress report).

Activity 8.3.1.4.2.2.3 - Borehole evaluation of faults and fractures. The objectives of this activity are to assess the reliability and usefulness of available borehole techniques for identifying and characterizing the subsurface fracture distribution; to determine vertical and lateral variability and characteristics of subsurface fractures; and to identify subsurface characteristics of fault zones.

Work under this activity is being conducted in conjunction with the surface-fracture network studies, Activity 8.3.1.4.2.2.2.

Activity 8.3.1.4.2.2.4 - Geologic mapping of the Exploratory Studies Facility. The objectives of this activity are to determine the vertical and horizontal variability of fracture networks in the ESF ramps, drifts, and boreholes; to characterize major faults and fault zones in the subsurface; to map the lithostratigraphic features of the subunits and the abundance and character of lithophysal zones; and to assist in the evaluation of test locations in the ESF.

Mapping continued in the ESF south ramp, the Thermal Testing Facility, and the Northern Ghost Dance Fault Alcove. Progress for the individual components of the underground mapping are as follows:

South Ramp:

- Stereophotography completed to Station 72 + 08 (7208 m)

- Full-periphery geologic maps completed to Station 72 + 10 (7210 m)

- Detailed line surveys completed to Station 72 + 07 (7207 m)

- Calculation of Rock Quality Designation (RQD) completed to Station 71 + 22 (7122 m)

- Calculation of rock-quality (Q) and rock-mass-rating (RMR) coefficient completed to Station 70 + 55 (7055 m).

Thermal Test Facility:

- Stereophotography, full-periphery geologic maps, and detailed line surveys completed within the facility.

Northern Ghost Dance Fault Alcove:

- Stereophotography, full-periphery geologic maps, and detailed line surveys completed to Station 1 + 30 (130 m)

Geology of the North Ramp—Station 4 + 00 to 28 + 00. A report has been completed that describes the results of structural and stratigraphic studies of the various lithologies exposed in the ESF north ramp (D. L. Barr et al., 1996). The rock units penetrated by the ESF tunnel along the north ramp include the pre-Rainier Mesa Tuff, Tuff "x," Tiva Canyon Tuff, Yucca Mountain Tuff and Pah Canyon Tuff and associated bedded tuffs, upper Topopah Spring Tuff, and the crystal-rich, middle nonlithophysal zone of the Topopah Spring Tuff, which is the host rock for the potential repository. Included in the report are summary descriptions of lithostratigraphic units, statistical analyses of 3735 fractures measured during geologic mapping, detailed line surveying, an analysis of the geotechnical and engineering characteristics of the tunnel, Upper Paintbrush Tuff Contact Alcove, and the Lower Paintbrush Tuff Contact Alcove.

The exposures in the north ramp, Upper Paintbrush Contact Alcove, and Lower Paintbrush Contact Alcove clarified and confirmed several lithostratigraphic relationships that were previously inferred from studies of drill core, downhole video, and hydrologic properties. In particular, intervals of intense vapor-phase alteration, unusually large lithophysal cavities, and quartz-latite pumice blocks that were described in borehole logs of the Topopah Spring Tuff were shown to have stratigraphic significance in north-ramp exposures (D. L. Barr et al., 1996). The report outlined a preliminary geologic history for a localized zone of degassing and intense vapor-phase alteration that was exposed at the top of the pumice-fall deposits that overlie the Topopah Spring ignimbrite. Exposures of the disrupted distal edge of the Yucca Mountain Tuff and the zeolitically altered lower portion of the Pah Canyon Tuff were also described.

The fracture network from Station 4 + 00 (400 m) to Station 21 + 87 (2187 m) in the north ramp was divided into three structural sets using cluster analysis. The first set was composed of east-west-striking cooling fractures that occur primarily in the Tiva Canyon Tuff and Topopah Spring Tuff. A second set, which occurs in all geologic units, had fractures with north-south strikes and steep, westward dips that represent the dominant tectonic fracture direction. A third set of fractures comprised shallowly dipping, subhorizontal features, such as vapor-phase partings, that occur mostly in the Tiva Canyon Tuff and Topopah Spring Tuff. These fracture sets were less clearly defined from Station 21 + 87 (2187 m) to Station 28 + 00 (2800 m) where the tunnel orientation changes in the turn from the north ramp to the main drift.

Maximum-variance, principal-component analysis of the detailed line survey data set indicated that fracture length and/or maximum aperture could be used to characterize these tuff units. Factor scores were used to divide the tunnel into eight distinct zones, seven of which correspond to lithologic contacts or welding breaks. The analysis less clearly defined four additional boundaries, two of which correspond to lithologic contacts and one that occurred at a break in vapor-phase alteration.

Geotechnical characterization of the ESF north ramp focused primarily on rock-mass quality and rock-mass mechanical properties. Descriptions are based on two empirical rock-mass classification systems, rock quality (Q system) and rock-mass rating. Data on rock quality (Q) and rock-mass rating were used to divide the north ramp of the ESF into ten sections of variable length, each of which was defined using geotechnical characteristics. The section with the highest average ratings is located from Station 7 + 95 (795 m) to Station 8 + 80 (880 m). This section, which consists of the nonwelded to partly welded, crystal-poor vitric zone of the Tiva Canyon Tuff and underlying pre-Tiva Canyon Tuff bedded tuff, has an average Q value of 30 and an average rock-mass rating value of 75, both of which are designated as "good." The section with the poorest average ratings is located from Station 10 + 70 (1070 m) to Station 11 + 80 (1180 m). This section, which consists of a faulted zone of the Topopah Spring Tuff crystal-rich vitrophyre, has an average Q value of 1.2 and an average rock-mass rating value of 55, which were designated as poor and fair, respectively.

Geology of the Main Drift—Station 28 + 00 to 55 + 00. A report (Albin et al., 1997) describes the results of structural and stratigraphic studies of the ESF main drift. The main drift is excavated almost entirely within the Tptpmn with only small exposures of the underlying Tptpll beyond Station 53 + 00 (5300 m). The comparison of the pre-construction geologic cross section of the main drift and the as-built geologic cross section shows strong agreement between the geology predicted and that encountered. The discontinuities (faults, joints, shears, and fractures) were divided into four sets on the basis of orientation, with a significant number of random orientations. Three of the sets occur throughout the ESF main drift, but the fourth set occurs only between Station 28 + 00 (2800 m) and Station 37 + 00 (3700 m). Set 1 was by far the most prominent, consisting of discontinuities striking generally between 100° and 150° and dipping 70° or more southwest. Set 2 consisted of discontinuities striking between 200° and 230° and dipping 70° or more northwest. Set 3 consisted of discontinuities striking between 280° and 320° and dipping less than 40° northeast. Set 4 consisted of discontinuities striking between 270° to 330° (similar to Sets 1 and 3), and dipping between 40° to 60° northeast (intermediate between Sets 1 and 2).

The ESF main drift was divided into four domains on the basis of structural characteristics. The first domain, extending from Station 28 + 00 (2800 m) to Station 37 + 00 (3700 m), was the only domain in which Set 4 fractures were found in significant numbers. In the second domain, Sets 1, 2, and 3 were well defined with relatively few random fractures. The third domain was defined by the intensely fractured zone that extends from Station 42 + 00 (4200 m) and to Station 51 + 50 (5150 m) and is dominated by Set 1 fractures. The fourth domain, from Station 51 + 50 (5150 m) to Station 55 + 00 (5500 m), consists predominantly of Set 1 and Set 2 fractures and has a high density of Set 1 and Set 2 faults and shears. Cluster analysis (using the computer program Clustran) also was performed and resulted in four sets of discontinuities. Three of the sets were in general agreement with the sets identified through other analytical methods. The fourth set (Set 4) was identified by cluster analysis. The two main differences between the cluster analysis and the other methods used are: (1) Clustran groups all the discontinuities into the four sets without a "random" category and (2) the sets identified by cluster analysis include a wider range of orientations both in terms of strike and dip.

Maximum-variance, principal-component analysis also was performed on the main drift detailed line survey data. The analysis indicated that the most useful parameters for characterizing the crystal-poor, middle nonlithophysal zone of the Topopah Spring Tuff (repository horizon) are maximum aperture, followed by infilling thickness and fracture length. A two-factor solution was obtained. Factor 1 scores are a function of infilling thickness, maximum aperture, and fracture length. Factor 2 scores are a function of fracture dip and minimum aperture. Both factor 1 and 2 scores were used to identify significant structural heterogeneities within the ESF main drift. Statistical correlations between strike and factor scores identified strike ranges with similar characteristics.

Geotechnical characterization of the main drift focused primarily on rock-mass quality and rock-mass mechanical properties. The averages of all the rating systems yield ratings of poor to fair for all of the main drift. The average rock quality ratings are fair in the first domain, and then generally decrease to poor southward in the main drift.

Activity 8.3.1.4.2.2.5 - Seismic tomography/vertical seismic profiling. The objectives of this activity are to investigate, and if successful, provide a means for broadly detecting and characterizing the subsurface fracture network in regions between the surface, boreholes, and underground workings; and to calibrate and relate the seismic propagation characteristics of the host rock to the fracture patterns observed in boreholes and underground workings, and extrapolate the observed fracture patterns to the surrounding region.

Interpretation of the amount, style, depth, and continuity of faulting in the proposed repository volume were continued using previously collected geophysical data. The most closely examined area was in the vicinity of the Ghost Dance fault. Evaluation of data from vertical seismic profile, surface seismic, gravity, ground magnetic, and magnetotelluric surveys (Majer et al., 1996) continued. Continued interpretation of the high resolution seismic data (Majer et al., 1996) in the repository region identified no seismic reflections from the interface between Paleozoic and Tertiary units. The amount of offset of the Paleozoic-Cenozoic unconformity remains indeterminate, since geologic control from drilling has not been established. Furthermore, the energy sources for the shallow seismic work were not strong enough to unequivocally resolve such a deep feature. In addition, three-dimensional gravity models remain consistent with minimal offset of the unconformity beneath the repository block (comparable to surface offsets of a few meters).

The results of the regional geophysical surveys (Brocher et al., 1996) continue to indicate that in this area the faulting is a classic example of steeply dipping Basin and Range extensional faulting with cross-cutting faults intersecting the normal faulting. Interpretation of data from regional seismic reflection studies were previously interpreted [see Section 3.3.3 of Progress Report #15 (DOE, 1997e)] as a reflector corresponding to an offset of the unconformity of about 1500 m by the Ghost Dance fault. In a meeting of the scientists on December 16, 1996, at the Lawrence Berkeley National Laboratories, it was agreed that the possible escarpment interpreted in the Paleozoic unconformity under Yucca Mountain could not possibly be the Ghost Dance fault. The orientation and geometry of this feature are not compatible with that of the Ghost Dance fault.

Interpretation of data from surface and borehole velocity studies across Yucca Mountain continue to indicate that in addition to local heterogeneity, a general trend of increasing seismic velocity exists from north to south This trend continues to imply increasing porosity to the north.

Forecast: Scheduled work includes the initial field work, compilation of the linework gained by the field work, digitization of the lines and symbols (strike and dips, lineations, breccia symbols, etc.), development of the underlying topographic coverage, and preparing a report that contains an overview of the geology and structural history of Yucca Mountain and rock-unit descriptions. Geologic mapping of the remaining part of the site area at 1:24,000-scale probably will be completed by May 1997 The cartography, symbols, digitization, and compilation of the topographic coverages for the geologic map probably will be completed by mid-June; writing of the report will begin in April

Geophysical work during the remainder of FY 1997 will focus on synthesis and modeling. This work will include the depth-to-magnetic-basement work and the final evaluation of the surface seismic data from both the repository and regional lines (Majer et al., 1996; Brocher et al., 1996, respectively). The significance of alternative interpretations of the amount of offset of the Paleozoic surface at depth will be further evaluated. In particular, the need for additional control from deep drilling will be evaluated. Documentation will be complete in April 1997. The results will be discussed in Progress Report #17. Beginning in the next reporting period.

seismic tomography data that have already been collected from UE-25 UZ#16 will be reprocessed.

In fracture studies, evaluation of the structural significance of sample locations in the ESF that show chlorine-36 from weapons testing will continue (see Section 3.1.5 of this progress report). Activities will include evaluating sample localities along the south ramp of the ESF, evaluating possible influence of fracture orientation on preferred flow paths, and further evaluation of the interplay between structural features and spatially distributed percolation.

In the ESF mapping activity, Project geologists expect to accomplish the following during the second half of FY 1997: (1) complete mapping of the ESF south ramp including full-periphery geologic mapping, detailed line survey, stereophotography, and rock-mass classification; and (2) complete mapping of the Northern and Southern Ghost Dance Fault Alcoves. In addition, a report will be prepared on the geology of the ESF south ramp, including results of stratigraphic and structural studies similar to those documented for the ESF north ramp and main drift.

3.3.5 Study 8.3.1.4.2.3 - Three-Dimensional Geologic Model

The objective of this study is to develop a three-dimensional geologic model of the site area to serve as a framework for subsequent rock properties, mineralogic, hydrologic, flow, and transport models. Additionally, rock properties and mineralogic models are integrated into the geologic framework to form an Integrated Site Model (ISM). Model development involves synthesis of the results of other geologic and geophysical studies.

The three-dimensional geologic framework and integrated site model of Yucca Mountain, version ISM2.0 was completed along with a report (CRWMS M&O, 1997a). The geologic framework used the results of geologic mapping; borehole and outcrop data including lithologic logs, measured sections, and geologic maps; surface geophysics; borehole geophysics; and ESF studies. Lithostratigraphic horizons included in the framework were determined from discussions with model users, including those modeling rock properties, unsaturated zone hydrology, mineralogy, radionuclide transport, and performance assessment. Forty-four faults were included in ISM2.0. All fault geometries (including dips) were formulated through extensive workshops and iteration with geologists. The framework model covers the site area from the topographic surface down to the top of the Paleozoic section and includes 34 lithostratigraphic horizons. The top of the Paleozoic was selected after evaluation of two interpretations: those of Majer et al. (1996) and Brocher et al. (1996). Majer's interpretation was selected based on its ability to provide the best fit to various geologic and geophysical data and was considered detailed enough to meet the needs of the users of the model. Three-dimensional mineralogic and rock properties models were integrated into the geologic framework to form the integrated site model. Modeled rock properties include bulk porosity, thermal conductivity, matrix porosity, density, and hydraulic conductivity in the Tiva Canyon Tuff, Yucca Mountain Tuff, Pah Canyon Tuff, Topopah Spring Tuff, Calico Hills Formation, and Prow Pass Tuff (see Section 3.3.6). Modeled mineral abundances include zeolites, crystobalite, trydimite, and smectite.

Forecast: Work will continue to support model users. The three-dimensional integrated site model (version ISM3.0) will be developed. Additional geophysical data will be synthesized to be incorporated in the geologic framework, including seismic reflection profiles and magnetic interpretations. Work will focus on supporting other modeling activities, providing model products for use in management decisions and other Project activities, and compiling new information and revising old information to construct version ISM3.0. Key elements to be incorporated into version ISM3.0 include the site area geologic map, final ESF geologic mapping, and revised borehole lithostratigraphic contacts. New boreholes will be used both to evaluate version ISM2.0 and to construct version ISM3.0.

3.3.6 Study 8.3.1.4.3.1 - Systematic Acquisition of Site-Specific Subsurface Information

The objective of this study is to systematically acquire physical rock samples, analytical data, and basic descriptions of the subsurface geology of the repository site. These samples and information are important for characterizing the three-dimensional distribution of rock characteristics, and hydrologic and geochemical variables, for the unsaturated zone at Yucca Mountain. Only one activity is planned under this study.

Activity 8.3.1.4.3.1.1 - Systematic Drilling Program. The objective of the Systematic Drilling Program is to acquire rock samples, analytical data, and basic descriptions of the subsurface geology of the potential repository block for characterizing and evaluating the three-dimensional distribution of rock characteristics, and hydrological and geochemical variables. Core samples taken from selected drillholes provide information related to the design of the exploratory studies facility main test level and relevant geologic information required for understanding the deeper portions of the repository block.

No drilling or other field-based site characterization activities were conducted under this study during the reporting period.

Two comprehensive data reports describing the geology of the USW SD-7 (Rautman and Engstrom, 1996a), and USW SD-12 (Rautman and Engstrom, 1996b) drillholes were published and issued. These reports describe (a) geology, quantitative and semiquantitative information on fractures, lithophysae, core recovery, and rock-quality measurements; (b) framework bulk and hydrologic properties; (c) geophysical well logs; and (d) x-ray diffraction mineralogic data. The report for drillhole USW SD-9 (Engstrom and Rautman, 1996) has been revised to incorporate minor editorial and format changes. The content of this drillhole report is the same as that for the reports on drillholes USW SD-7 and USW SD-12 (Rautman and Engstrom, 1996a and b).

Work products were provided to support several data synthesis reports and performance assessment analyses.

Forecast: Borehole USW SD-6 is planned to be drilled during the next fiscal year. Additional data collected will be integrated into the assessment of the Systematic Drilling Program.

3.3.7 Study 8.3.1.4.3.2 - Three-Dimensional Rock Characteristics Models

The objective of this study is to produce numerical models of rock properties for use in various ESF design-evaluation and performance-assessment analyses, principally using geostatistical and other computer modeling methods. The study also will support the development of new computer algorithms and computer software required to accomplish the modeling.

Activity 8.3.1.4.3.2.1 - Development of three-dimensional models of rock characteristics at the repository site. The objective of this activity is to develop computer-based three-dimensional models that integrate quantitative and semiquantitative data on rock characteristics in light of constraining information developed by studies of the geologic framework of the Yucca Mountain site.

The rock properties models described in Progress Report #15 (DOE, 1997e) and incorporated into the Integrated Site Model, version 1.0, have been completely regenerated for ISM 2.0 (see Section 3.3.5 of this progress report) to incorporate additional conditioning data not previously available. In addition, secondary property models were generated using the expanded data base, and a modestly revised version of the linear coregionalization algorithm. The following secondary property models are now available for the entire Yucca Mountain extended site area:

For the upper PTn model unit (nominal 250 × 250 × 2-m grid spacing):

- porosity
- bulk density
- saturated hydraulic conductivity.

For the TSw model unit (including the repository horizon; nominal 250 × 250 × 10-m grid spacing):

- matrix porosity (excludes effect of lithophysae)
- lithophysal porosity (includes effect of lithophysae where present)
- bulk density
- thermal conductivity
- saturated hydraulic conductivity.

For the combined Calico Hills-Prow Pass (CH-PP) model unit (nominal 250 × 250 × 10-m grid spacing):

- porosity
- bulk density
- alteration category ("zeolitic" versus unaltered)
- saturated hydraulic conductivity.

Uncertainty estimates were also generated this fiscal year for all material properties models (except bulk density), and these were included in the ISM 2.0 release.

The new data incorporated in the ISM 2.0 release models were essentially all computed porosity values derived from processing borehole geophysical measurements. Geophysical logs from almost all "older" (pre-1986) drillholes have now been converted to computed porosity values (Nelson, 1996); the holes included were the G- and H-series (8 drillholes) in addition to the previously available WT-series holes. Computed porosity values are now also available for the entire suite of "modern" (site characterization) boreholes (CRWMS M&O, 1996h).

Several "old" drillholes (USW G-2, USW WT-2, USW WT-10, and USW WT-12) were relogged using modern geophysical tools. The data from relogging provided some quantitative correlation of porosity values between logging methods. Other new data consist of direct mineralogic indicators of zeolite alteration incorporated into the indicator modeling of rock type for the Calico Hills-Prow Pass combined stratigraphic unit (Chipera et al., 1996).

The creation of the rock property models for ISM 2.0 followed essentially the same process outlined in Progress Report #15 (DOE, 1997e). Porosity data were compiled, and the measured values were associated with stratigraphic coordinates for each of the three model units. "Stratigraphic coordinates" attempt to place the conditioning data values back into their original depositional position within a rock unit by measuring positions relative to the base (or top) of the model unit. Differences in "total" porosity and "water-filled" porosity were used to identify intervals within the Calico Hills-Prow Pass stratigraphic interval that are likely to have been zeolitized; these altered intervals were coded using indicator flags (ones and zeros). Horizon-specific mineralogic data for drillholes USW SD-7, USW SD-9, and USW SD-12 (Chipera et al., 1996) were incorporated directly at the appropriate stratigraphic elevations.

Spatial correlation patterns were then quantified by computing experimental variograms (separately for each model unit) in different directions. Spatial correlation for porosity is strongly anisotropic vertically, as is expected for layered rock sequences. Nested three-dimensional anisotropic variogram models exhibited at least two, and more frequently three, different ranges of spatial continuity. Correlation patterns for the altered-unaltered indicator categories are also strongly anisotropic.

These variogram models and the conditioning data served as input to the geostatistical modeling algorithms. Sequential gaussian simulation was used to generate 100 replicate, statistically indistinguishable, plausible models of porosity for each of the three model units. These simulated models reproduce the measured data at data locations (subject to discretization limits), the overall distribution of relevant porosity values (histograms), and approximate the variograms exhibited by the data. Where there were departures of the input modeled variogram from the variogram of the completed property model, the departures were observed to be more like the original experimental spatial continuity pattern. Modeling of the zeolitic and non-zeolitic lithologies within the Calico Hills-Prow Pass combined interval used sequential indicator simulation in the categorical mode.

Forecast: Most of the remainder of FY 1997 will be dedicated to completing and closing out the scientific notebook associated with the modeling for ISM 2.0. A comprehensive report describing the modeling activities, the data used, and the resulting three-dimensional models is currently being prepared.

Future rock-properties modeling in support of revisions to ISM 2.0 will most likely be focused on specific problems, rather than generalized revisions to the gross distribution of properties throughout the extended site area. Specifically, evaluation of porosity data from the PTn model unit during the data compilation stage early in the reporting period identified the presence of some form of hydrous-mineral alteration at particular stratigraphic levels within the PTn. Given the high relative structural position of the PTn unit in most locations with respect to the present and former water tables, it seems most likely that this hydrous-phase alteration involves montmorillonitic clays, rather than zeolites as in the Calico Hills-Prow Pass interval. Although the presence of clay alteration has been noted previously in and adjacent to this stratigraphic interval (e.g., Buesch et al., 1996b), the influence of such alteration on hydrologic properties in predictive modeling has not been addressed systematically.

A second topic of focused modeling activity is likely to be the production of a quantitative model of fracturing within the site. Fractures are important both as potential fast flow paths and as flow barriers within the unsaturated zone because of capillary effects between matrix pores and the "pores" formed by fracture apertures. Other modeling will likely be of smaller volumetric scope and related directly to specific performance assessment needs.

3.4 CLIMATE (SCP SECTION 8.3.1.5)

Changes to the Climate Program since the SCP was issued are summarized in Appendix A, Section A.1.4.

3.4.1 Study 8.3.1.5.1.1 - Characterization of Modern Regional Climate

The objective of this study is to provide a baseline for an analysis and interpretation of isotopic data from modern precipitation to provide an understanding of its seasonal and spatial variability.

No progress was made during the reporting period; this was an unfunded study.

Forecast: No activity is planned for this study during FY 1997.

3.4.2 Study 8.3.1.5.1.2 - Paleoclimate Study: Lake, Playa, and Marsh Deposits

The objective of this study is to establish the nature, duration, and amplitude of paleoclimate changes in the Yucca Mountain area, based on paleontologic, geochemical, stratigraphic, sedimentologic, and geochronological data obtained from lacustrine, playa, and marsh sediments in or near southern Nevada.

Activity-level descriptions for this study have been combined because they are so closely interrelated. The results of this study are being integrated with the results of Study 8.3.1.5.1.3 (Section 3.4.3 of this progress report). Results of both studies are being synthesized under Study 8.3.1.5.1.5 (Section 3.4.5 of this progress report).

The higher resolution analysis of Owens Lake diatom and ostracode stratigraphy at Owens Lake, California, was nearly completed. Approximately 450 new diatom samples from about 1000 total and about 600 new ostracode samples from about 1400 total have been added to the data set. Paleolimnologic and paleoclimatic data from this record now extend back to 500 ka, with a 500-year resolution between 200 and 0 ka and a 750-year resolution between 500 and 200 ka. The results from both the diatom and ostracode high resolution records indicate that limno-climatic changes at Owens Lake were extremely rapid; the lake often varied between a full and overflowing, freshwater system and a closed, saline or even dry system in less than a millennium. Such short-term, rapid changes are more characteristic of interglacial and transitional periods than of glacial periods. Core and sample coverage exists to further refine and document the transition time between specific climate modes at Owens Lake.

The distribution of freshwater planktic diatoms, which prosper in the late summer, have been interpreted to indicate increased late summer precipitation in the Owens Lake area (Bradbury, in press). Summer precipitation preferentially occurred during transitions between glacial and interglacial periods, presumably when maximal summer insolation produced strong thermal gradients between the Subtropical High and Great Basin Low to deliver precipitation from the eastern Pacific Ocean to the region. Summer precipitation in the Owens Lake and Yucca Mountain area may have distinctive geomorphic and infiltration consequences compared with past intervals of increased winter precipitation.

The principal new ostracode data come from samples of cores at the top of the stratigraphic section. The distribution of ostracodes in those cores dating from about 55 ka through the Holocene (10 ka to present) show very rapid (decadal) shifts from dry climates like those of today to brief periods of warm but wet climates probably supported by summer rain. The appearance of full glacial climate (25 ka) was also rapid, but this condition persisted for several millennia. Full glacial climate was characterized by precipitation levels well above those of today and by mean annual air temperatures well below those of today.

The aquatic microfossil data from Owens Lake and elsewhere are now being integrated into climate scenarios. Discussions with climatologists from Scripps Institute of Oceanography, La Jolla, California, have provided modern climate scenarios that might be partial modern analogs for summer precipitation regimes in the Owens Lake area. Information also has been assembled that provides insights into surface- and ground-water sources available for the lake in different climates. The integration of aquatic microfossil records of past seasonal climatic conditions with site specific records of past vegetation [packrat (*Neotoma sp.*) middens] and past hydrology (paleodischarge deposits) will document a broader range of paleoenvironmental scenarios for Yucca Mountain. These scenarios will provide a basis for forecasting future climate and actual boundary conditions suited for input to the hydrologic models designed to evaluate the performance of the potential repository.

Comparison of the Owens Lake record with other lacustrine records has supported and extended paleoclimatic interpretations for the Great Basin. For example, a comparison of glacial and interglacial climates at Owens Lake (36°N) with those at Tule Lake (42°N) indicates that wet, glacial climates at Owens Lake correspond to dry glacial climates at Tule Lake, reflecting the southern displacement of storm tracks by the Cordilleran and Laurentide ice sheets. Interglacial climates at these sites are also antithetical; dry at Owens Lake, but wet at Tule Lake (Smith et al., 1997). This comparison helps to place the Owens Lake record within the context of hemispheric climate change that has been explored by global climate models and the integration of paleoclimate information derived from pollen analysis in western North America.

A comparison of the Owens Lake record with the uranium-series-dated record at Death Valley, California, shows correlative changes of the past hydrologic balance at both sites. This correlation helps support the extrapolated Owens Lake chronology: a long and very wet penultimate glacial period and increased summer precipitation during periods of transition between glacial and interglacial climates. During the penultimate glacial period, both Owens Lake and Death Valley contained large lakes that persisted for much of about 40 k years, punctuated by drier periods of one to three millennia. Surprisingly, lakes persisted in both Death Valley and at Owens Lake at the beginning of the last interglacial (130 ka) when marine records suggest minimal glacial ice (less than today) and high sea-level stands. The persistence of lacustrine conditions in the southern Great Basin in early interglacial climates may reflect summer precipitation derived from the eastern Pacific Ocean. The new data suggest that in the last interglacial summer precipitation was extensive enough to both maintain runoff and become recharge

During the reporting period, ostracode data collection was completed for sites in the Las Vegas and the Indian Springs Valleys. The existing data set is composed of about 650 samples from three principal stratigraphic sections. Ostracode subsamples are now being collected for stable isotope analyzes. Existing stable-isotope and ostracode data suggest that the wetlands were supported by winter precipitation and probably were through-flow systems. The kinds of ostracodes found in deposits indicate that spring discharge emanated from both the alluvial and regional-carbonate aquifers near Corn Creek Flats in the upper Las Vegas Valley. Discharge appears to have come largely, but not exclusively, from alluvial aquifers in the Indian Springs Valley.

Radiocarbon-age control shows that deposition in the pale[illegible]etlands was more episodic than originally believed. Although preliminary, the current interpretations at low elevation in packrat (*Neotoma sp.*) middens suggest a correspondence of sediment accumulation and the existence of white fir (*Abies concolor*), which is believed to identify the wettest episodes during the last glacial period (40 to 10 ka). This would imply that the local water-table elevation only rose above the surface of the valley bottom during century- to millennium long wet periods

The Las Vegas Valley deposits, specifically those from Corn Creek Flats, appear to contain relatively old, possibly penultimate glacial (170 to 130 ka) sediments. The Owens Lake record, and now the paleowetland records, indicate that the penultimate glacial period was substantially wetter than the last glacial period (40 to 10 ka).

Activity 8.3.1.5.1.2.1 - Paleontologic analyses. The objective of this activity is to assemble and interpret, in paleoclimate terms, detailed fossil records from ostracodes, diatoms, and pollen, along with other types of fossils as warranted by specific paleoclimate questions.

See the discussion under the heading for Study 8.3.1.5.1.2.

Activity 8.3.1.5.1.2.2 - Analysis of the stratigraphy-sedimentology of marsh, lacustrine, and playa deposits. The objectives of this activity are to identify and characterize the general physical and chemical properties of sedimentary units from outcrops, shore deposits, and cores (providing a physical and relative temporal framework for various paleoenvironmental studies), and to determine the specific environment of deposition for the sedimentary units using the principles of clastic and chemical sedimentology.

See the discussion under the heading for Study 8.3.1.5.1.2.

Activity 8.3.1.5.1.2.3 - Geochemical analyses of lake, marsh, and playa deposits. The objective of this activity is to assemble and interpret, in paleoclimate terms, detailed records of stable isotopic, trace metal, and mineralogical data. The resulting interpretations provide the climate framework within which Yucca Mountain infiltration, percolation, recharge, and water-table elevation may be understood in climate terms.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.5.1.2.4 - Chronologic analyses of lake, playa, and marsh deposits. The objective of this activity is to obtain an accurate chronologic framework for the paleoclimatic information acquired in this study. All age information should, whenever possible, be tested with other techniques to reduce uncertainties.

See the discussion under the heading for Study 8.3.1.5.1.2.

Forecast: Closer-interval sampling and analysis of the Owens Lake and other data from other regional and local climate records will be required to resolve the timing, rate, and magnitude issues of climate change. Data sets will be aligned with each other along a time axis and then integrated to obtain some composite perspective of past climate change. Time permitting, ostracode subsamples will be collected from the Owens Lake record for stable isotope analyzes. Collection will continue of ostracode subsamples for stable isotope analyses from the wetland deposits at Corn Creek Flats in the Las Vegas and Indian Springs Valleys.

3.4.3 Study 8.3.1.5.1.3 - Climatic Implications of Terrestrial Paleoecology

The objectives of this study are to provide quantitative estimates of changes in climatic variables (e.g., precipitation and temperature) for the southern Great Basin, develop transfer functions, response surfaces, or both by statistically comparing modern climate to the vegetation data; and reconstruct climates from the paleovegetation data using these equations.

Activity 8.3.1.5.1.3.1 - Analysis of packrat middens. The objectives of this activity are to use radiocarbon dated macrofossils of climate indicator plant species to (1) identify periods of greatest late Quaternary precipitation, (2) determine durations of these periods, and (3) determine the rates of onset and demise of these periods.

No progress was made during the reporting period; this was an unfunded activity. Remaining statistical analysis and integration of data are being conducted under Study 8.3.1.5.1.5 (Section 3.4.5 of this progress report).

Activity 8.3.1.5.1.3.2 - Analysis of pollen samples. The objectives of this activity are to use high frequency pollen records from southern Nevada to (1) determine the frequencies, rates and magnitudes of current Holocene climatic change, and (2) provide baseline data for the generation of analogs or transfer functions of high resolution climate change for the interpretation of less complete Pleistocene paleoclimatic proxy records.

No progress was made during the reporting period; this was an unfunded activity. Remaining analysis and integration of data are being conducted under Study 8.3.1.5.1.5 (Section 3.4.5 of this progress report).

Activity 8.3.1.5.1.3.3 - Determination of vegetation-climate relationships. The objective of this activity is to translate the vegetational records provided by packrat midden and palynological investigations and available dendroclimatalogical data into qualitative estimates of past climatic variables.

No progress was made during the reporting period; this was an unfunded activity. Remaining statistical analysis and integration of data are being conducted under Study 8.3.1.5.1.5 (Section 3.4.5 of this progress report).

Forecast: This study was not funded in FY 1997. Remaining interpretation and integration of data are being conducted under Study 8.3.1.5.1.5 (Section 3.4.5 of this progress report).

3.4.4 Study 8.3.1.5.1.4 - Analysis of the Paleoenvironmental History of the Yucca Mountain Region

The objectives of this study are to evaluate the paleoenvironmental record at Yucca Mountain and surroundings in light of inferred paleoclimate history of the southern Great Basin; to provide information to distinguish the effects of surficial processes from those of tectonic activity, based on the character and distribution of surficial deposits; and to evaluate the age of tectonic events

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In situ cosmogenic beryllium dating techniques have been used to demonstrated that the vertical bedrock erosion rate on the climatically insensitive and erosionally resistant welded tuffs of Yucca Mountain is approximately 1 m per million years. This is less than half of the erosion rate of unconsolidated materials on the hillslopes of Yucca Mountain. This erosion rate on the resistant bedrock ridges of the mountain supports the contention that erosion proceeded very slowly by weathering of large blocks of tuff during the colder, wetter climatic periods in the Quaternary. Additional data on the timing of the deposition and later incisement of the alluvial fills in Fortymile Wash indicate that deposition of the material forming the higher terrace probably occurred between 200 to 250 ka during oxygen isotope stage 7, and the final incision of these alluvial materials began about 150 ka ago during the colder and much wetter climate of oxygen isotope stage 6. (For depiction of oxygen stages, see for example, Forester et al., 1996a Figures 8 and 9.)

Activity 8.3.1.5.1.4.1 - Modeling of soil properties in the Yucca Mountain region. The objectives of this activity are (a) to determine the relations among properties of late Holocene soils and climatic parameters; (b) to compare properties of selected soils at Poohed Mesa and near Tonopah that formed under conditions similar to those that may have existed at Yucca Mountain during Pleistocene pluvial conditions; (c) to compare postulated past climates based on properties of early Holocene and Pleistocene soils to paleoclimatic models that are reconstructed from other lines of evidence, such as paleolimnology and terrestrial paleoecology, as a check on these models; (d) to postulate past climates based on the depth, distribution, and quantity of pedogenic carbonate and other soil parameters; and (e) to quantify rates of soil development in specific climates for use as a dating tool for Quaternary deposits and ages of fault movements.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.5.1.4.2 - Surficial deposits mapping of the Yucca Mountain area. The objectives of this activity are to determine the distribution, age, genesis, soil properties, and physical properties of surficial deposits in the Yucca Mountain area; to evaluate the influences of climate and tectonics on the genesis of surficial deposits; to provide a map of surficial deposits for facility placement planning, geomorphic studies, tectonics studies, engineering property studies, and surface infiltration studies; and to determine the distribution of major concentrations of calcite-silica deposits at or near the ground surface at Yucca Mountain.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.5.1.4.3 - Eolian history of the Yucca Mountain region. The objectives of this activity are to document eolian erosion and deposition in the Yucca Mountain area during the last 750,000 years; to determine paleoenvironmental conditions during times of eolian deposition and intervening times of surface stability and soil formation; and to determine source areas of sand and silt.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No activity is planned for this study during FY 1997.

3.4.5 Study 8.3.1.5.1.5 - Paleoclimate-Paleoenvironmental Synthesis

The objective of this study is to compare the paleoclimate estimates from the various proxy data sets and provide data synthesis in the formats required for future climate and paleohydrology investigations.

Relevant paleoclimate data sets are being finalized and interpreted. Coherent patterns of the timing, duration, character, and magnitude of climate change in the southern Great Basin are emerging. One particularly important finding is the approximate correspondence of the isotope record at Devils Hole (about 45 km south of Yucca Mountain, Figure. 3-1.) with the paleoclimate records from Owens Lake, California. Further, both of these records correspond to the marine oxygen isotope records, which document global changes in the temperature and ice volume of the earth. While not perfect, the correspondence means the timing and rate of change of past climate in the Yucca Mountain area coincided with the large, cyclic changes in global climate throughout the Quaternary. Because global climate records are closely linked to astronomically derived changes in solar insolation, and future changes in solar insolation can be accurately determined, forecasting the timing of climate change in the Yucca Mountain area is possible. Similarly, because the magnitude of regional climate change within particular segments of the solar insolation cycle is beginning to be understood, it also is possible, in general terms, to forecast the magnitude of future climate change in the Yucca Mountain region. In particular, the present-day segment of the insolation cycle resembles that which occurred 400 thousand years ago Consequently, the characteristics of climate that happened about 400 thousand years ago and onward could reasonably be expected to recur, in some general way, in southern Nevada, and at Yucca Mountain, in the future.

A correspondence has also been recognized between the Owens Lake climate record and local records near Yucca Mountain, including data on packrat (*Neotoma sp.*) middens, paleodischarge sites, and wetland deposits. The correspondence supports the interpretation of local, cyclic climate changes above the possible repository and provides specific information about the magnitude of local climate change. Although the local hydrologic setting of Owens Lake ensures that it is and was generally wetter than Yucca Mountain, packrat midden and ostracode analogs indicate precipitation at Yucca Mountain was commonly two times greater than present when Owens Lake was fresh and overflowing during the last glacial period. Paleotemperature estimates from packrat middens near Yucca Mountain suggest mean annual temperature depressions of 5 to 10°C. This information has been correlated with diatom and ostracode assemblages and with sedimentological evidence of drop stones at Owens Lake that indicates that ice cover was present at Owens Lake throughout much of the winter during the last ice age. Uranium-series dates on terrace deposits also indicate incision at Fortymile Wash at the same time (30 to 20 ka), which implies increased vegetation cover and considerably higher winter precipitation. Spring discharge sites near Yucca Mountain became active as water tables rose and wetland communities of plants and animals flourished throughout southern Nevada.

A working group of paleoclimate specialists was formed to interpret and integrate paleoclimate proxy data sets and to interpolate those results to a grid system throughout the Yucca Mountain precipitation area. The group met during December and discussed the strengths and weaknesses of the principal fossil climate proxy groups, which are packrat midden data, pollen, diatoms, ostracodes, and mollusks. Although each fossil group records past climate in a different way and reveals different aspects of the climate system, there are remarkable similarities in the nature of past climate interpretations within the Yucca Mountain region. A common and important theme of the records of all proxy groups was that the last climate cycle was not a singular, wet, cold event. Instead, the period from about 40 to 10 ka was a complex sequence of climates ranging from very wet and only somewhat cooler than today (about 33 to 31 k radiocarbon years) to very dry, possibly drier than today, but very cold at the full glacial maximum (about 18 k radiocarbon years). Change in climate style often appears to have been very rapid, at decadal scales, and the duration of particular climate states ranged from a century to a millennial time frame.

The packrat midden data base has been compiled and placed in a single spread sheet for analysis. A key component of the analysis was the identification of where various combinations of tree and shrub species live today versus locations of middens, in the Yucca Mountain area, containing the same species. That information provides the basis for estimating past climate parameters and provides historic meteorological records from analog sites for use in modeling annual variability in past climate in the Yucca Mountain area. For example, the infiltration model developed by Flint et al. (in prep.) requires not just estimates of the magnitude of climate change, but also a measure of how that magnitude might have been expressed over several decades. A preliminary reconstruction for the wettest episode (33 to 31 ka) in the region during the period from 40 to 10 ka suggests the mean annual precipitation was about four times modern and the mean annual temperature was about 6°C cooler than today. That analysis did not include many of the smaller shrub species that, when accounted for, likely will lower the precipitation estimate and might decrease the temperature estimate.

__Forecast__: The second half of FY 1997 will be devoted to preparing reports on the paleolimnological and paleoclimatic changes from the Owens Lake record, as well as several papers dealing with marsh and wetland deposits. Past climate will be interpreted using packrat midden data for four different episodes in the period from 40 to 10 ka. The episodes range from very wet and cool to very dry and cold. Those estimates will then be interpolated along a grid system covering the region. The magnitude of climate change will be determined using a change in the elevations of the key plant taxa and compared with the analog estimations. The results of the packrat-midden analysis will be integrated with the diatom and ostracode work as new information becomes available.

3.4.6 __Study 8.3.1.5.1.6 - Characterization of the Future Regional Climate and Environments__

The objective of this study is to evaluate climate parameters for local Yucca Mountain scenarios over the next 100,000 years, with emphasis on the next 10,000 years for local climate scenarios that are both reasonably probable and relevant to repository performance.

__Activity 8.3.1.5.1.6.1 - Global climate modeling__. The objectives of this activity are to identify a set of global climate states (global boundary conditions) that are reasonably likely to occur over the period of concern, that are relevant to system performance, and that are pertinent to understanding global climate effects; and to use one or more carefully chosen global general circulation models with these selected global climate states to provide time-dependent boundary conditions for use as input to regional models.

With the selection and qualification of the modeling codes reported in Progress Report #13 (DOE, 1996f), and the preliminary identification of modeling states defined in the study plan, this activity is considered complete (see Progress Report #13, Section 3.4.6).

Activity 8.3.1.5.1.6.2 - Nested global-regional climate modeling. The objective of this activity is to embed higher-resolution regional climate modeling within global climate modeling to develop a capability to model future climate-induced conditions for use in modeling net future infiltration in the Yucca Mountain area.

Evaluation and validation of the nested climate model was completed, and this activity is considered complete. (See Appendix A, Section A.1.4.1 of this progress report.)

Activity 8.3.1.5.1.6.3 - Site-specific model output adjustment. The objective of this activity is to predict climatic parameters such as precipitation, soil moisture, and net infiltration rates for several possible future climate scenarios at Yucca Mountain.

The relevant aspects of this activity, originally planned as part of the future climate modeling study, are presently being addressed by the infiltration and unsaturated hydrologic modeling studies (Studies 8.3.1.2.2.1 and 8.3.1.2.2.9, Section 3.1.5 and Section 3.1.15, respectively, of this progress report). Activity 8.3.1.5.1.6.3 is therefore considered complete.

Activity 8.3.1.5.1.6.4 - Future climate synthesis. The objective of this activity is to analyze time-series data of climatic variability based on the paleoclimatic record to identify possible future scenarios of concern that may occur during the next 100,000 years.

The climate scenario selected for modeling during FY 1997 was a simulation using an atmosphere having six times the carbon dioxide concentration of the present-day atmosphere. This state represents the maximum possible future carbon dioxide concentration and represents an extreme state not seen in the paleoclimate record. Results of the simulation were submitted late in the reporting period (Thompson et al., 1997).

Forecast: A final summary report of the future climate modeling activity and results will be prepared for submittal by May 19, 1997

3.4.7 Study 8.3.1.5.2.1 - Characterization of the Quaternary Regional Hydrology

The objective of this study is to characterize the distribution of surface water, the unsaturated zone infiltration and percolation rates, and the ground-water potentiometric levels during the Quaternary in the vicinity of Yucca Mountain.

Symbols from lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpll and Tptpmn) are found in Buesch et al. (1996a).

Activity 8.3.1.5.2.1.1 - Regional paleoflood evaluation. The objectives of this activity are to identify the locations and investigate the hydraulic characteristics of paleoflood events and compare this evidence with the locations and characteristics of modern flooding and geomorphic processes; and to assess the character and severity of paleoflood and debris hazards and the potential of flood and debris hazards for the repository during the preclosure period.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.5.2.1.2 - Quaternary unsaturated zone hydrochemical analysis. The objectives of this activity are to determine the past infiltration and percolation history at Yucca Mountain by analyzing the isotopic and chemical characteristics of water from the unsaturated zone; and to understand the past unsaturated zone hydrologic system by modeling vadose water hydrochemistry to help predict the future hydrologic system.

The scope of this work was transferred to Activity 8.3.1.2.2.7.2 (Section 3.1.11 of this progress report).

Activity 8.3.1.5.2.1.3 - Evaluation of past discharge areas. The objectives of this activity are to determine the location, type, and extent of hydrogeologic units in the ground-water discharge areas of the Amargosa Desert and Death Valley; to understand the past quantity and quality of water in the discharge areas of Franklin Lake, Amargosa River, and Peter's Playa and to determine the paleohydrologic significance of Peter's Playa and Franklin Lake as discharge areas; to determine the location and hydrogeologic characteristics of paleospring deposits in the discharge area; to determine the location and amount of discharge by evapotranspiration that has occurred at past discharge sites; to understand the past and present discharge areas of the regional hydrologic system to predict the future saturated zone hydrologic system at Yucca Mountain; and to determine past ground-water levels in carbonate caverns as evidence of past hydrologic conditions.

Paleodischarge deposits in southern Crater Flat and the central Amargosa Valley were the focus of field study and sample collection. These deposits, within 15 to 20 km of Yucca Mountain, represent the nearest down-gradient discharge from the potential repository, and an understanding of the timing and volume of their past discharge and recharge sources is important to performance assessment. Project personnel sampled several stratigraphic sections 5 to 6 m in thickness from deposits in the Amargosa River Valley at the distal part of the Fortymile Wash fan (informally known as the Stateline deposits). In all instances, the sections consisted of fine-grained sediments (clays to fine sands) with variable amounts of authigenic carbonate or silica cementation. Additional samples were collected from discharge deposits along Highway 95 and in Crater Flat to verify their depositional ages and sources of discharging ground waters. To address these issues, 109 samples or subsamples were collected for geochronologic, stable and radiogenic isotopic, paleontologic, and geochemical study.

Preliminary data collected from these samples and the field investigations indicate the following:

1. The deposits making up the toe of the Fortymile fan record a variety of settings including marsh and lacustrine environments; fairly pure limestones that cap the sequence may have formed in spring-fed ponds in a setting similar to the modern-day Ash Meadows area. The relation between the Fortymile Wash and Amargosa fluvial systems and the absence of interbedded alluvial deposits in the section at the toe of the fan indicate that the wetlands generally maintained a positive topographic expression during the late Pleistocene.

2. The delta carbon-13 ($\delta^{13}C$) and delta oxygen-18 ($\delta^{18}O$) values of calcite cements and nodules from the Fortymile Wash fan deposits are compatible with spring discharge. The values are comparable to those from the Crater Flat and Highway 95 discharge sites, and both locations have some oxygen-18-enriched calcite that may indicate evaporative concentration of depositing waters.

3. Some of the samples from the Fortymile Wash fan contain water-soluble minerals. Such minerals might reflect episodes of drier climate during deposition of some of these sediments, or they might represent evaporative deposition from pore waters under the modern climate regime.

4. A 6-m-thick sequence of limestones and calcareous siltstones from near Scranton Well has $\delta^{13}C$ values that also are compatible with deposition from waters of spring discharge origin. Therefore, they might record a long history of ground-water discharge in the Amargosa Valley region.

5. Discharge deposits at the toe of the Fortymile Wash fan contain diatom assemblages similar to those at the Lathrop Wells diatomaceous deposit, the hydrochemistry and temperature of the discharging waters at the two locations also probably were similar.

6. Resampling and further thorium-230/uranium dating of sediments that bracket an apparent unconformity within the Lathrop Wells diatomaceous deposits verifies the age discontinuity between upper, diatom-rich units with ages less than 33 to 55 ka and underlying green sandy units with ages greater than ~130 ka.

7. Sediments from outlying outcrops of the Lathrop Wells diatomaceous deposit have diatom assemblages consistent with growth in the same waters that formed the main part of the deposit.

Regional-scale maps (1:250,000) of discharge sites active during the last full glacial (10 to 30 ka) and brief descriptions of the type of discharge system and sources of water were prepared in support of the three-dimensional, regional saturated zone hydrologic model (Section 3.1.4 of this progress report).

Studies of thermoluminescence dating of surface deposits near Yucca Mountain (Mahan et al., 1996) and Death Valley ostracode glacial-lake hydrochemical history (Forester et al., 1996b) reported the results of uranium-series dating, strontium isotopes, sedimentology, evaporite mineralogy, and ostracode assemblage studies. These studies indicated that during the last glacial Death Valley hosted a moderate-to-large lake supported by discharge from the Amargosa River. The maximum depth of that lake, probably about 90 m, was only maintained for a few centuries. Depths of a few tens of meters were more typical.

During the penultimate glacial, however, a much larger lake existed in Death Valley. The maximum depth of that lake would have placed its shorelines outside of Death Valley. That lake appears to have been largely supported by discharge from the Amargosa River, although some discharge may have come from the Owens River drainage. Hydrochemical constraints, as well as calcite delta carbon-13 values and diatom assemblages from the Tecopa Lake beds near Shoshone, California, indicate a significant contribution of ground-water discharge to the fluvial system.

Activity 8.3.1.5.2.1.4 - Analog recharge studies. The objective of this activity is to estimate the conditions and rates of ground-water recharge (infiltration) during the Quaternary Period in the vicinity of Yucca Mountain.

This activity was terminated in FY 1994; see Progress Reports #10 (DOE, 1994e) and #11 (DOE, 1995b) for details.

Activity 8.3.1.5.2.1.5 - Studies of calcite and opaline silica vein deposits. The objective of this activity is to determine the ages, distribution, origin, and paleohydrologic significance of calcite and opaline silica deposits along faults and fractures in the vicinity of Yucca Mountain.

Beginning in FY 1995, the emphasis of this activity shifted to determining the spatial and temporal distribution of flux through the repository block using isotopic ages and measured distributions of low-temperature calcite and opal fracture and cavity fillings that were deposited from downward percolating water and that are exposed in the ESF.

A report was submitted to DOE-YMSCO that summarized results to date of analysis of calcite and open fracture- and cavity-filling deposits in the ESF (Paces et al., 1996a). Radiocarbon ages of the low-temperature minerals range from 44 to 16 ka with the greatest number of ages distributed between 38 and 28 ka. Thorium-230/uranium ages, however, for subsamples from the same surface range from 28 ka to greater than 500 ka, with most between 50 and 400 ka. Both radiocarbon and thorium-230/uranium ages from secondary minerals near or within discrete zones of elevated chlorine-36 showed similar distributions relative to those in zones with background levels of chlorine-36. Oxygen-18, carbon-13, and strontium-87/strontium-86 values for the outermost calcites vary over limited ranges and reflect the compositions of modern calcite-rich soils and fractionation due to present geothermal gradients. However, data for the earliest calcite indicate that geothermal gradients may have been steeper, methane may have been the carbon species controlling carbon fractionation in the subsurface, and strontium was derived from a less-radiogenic source than the current soil reservoir. Assuming that all calcium was extracted from waters percolating through the unsaturated zone enroute to the water table, a minimum value of flux of about 2 mm/yr was obtained as an average over the last 12.7 million years. This value is considered provisional and is intended only to show the viability of this approach for calculating the temporal and spatial variability of past fluxes and possibly estimating future changes in flux. Surface records from the Yucca Mountain area indicate that the regional climate over the last million years was wetter and cooler relative to the present for as much as 80 percent of the time. However, there is no indication in the subsurface that deposition rates of calcite and opal varied greatly during this period even though higher waste tables throughout the region reflected greater recharge. These observations suggest that percolation through the Topopah Spring Tuff may have been buffered from variations in infiltration except in zones of high permeability and highest flux.

Samples of opal and calcite were collected from 28 sites in the ESF between Station 57 + 71 (5771 m) and Station 69 + 42 (6942 m). Relatively slow progress by the tunnel-boring machine during this reporting period limited the number of suitable sites available for sampling. In addition to sampling, line surveys to quantitatively assess the abundance of calcite and opal in fractures and cavities were completed between Station 35 + 00 (3500 m) and Station 69 + 30 (6930 m). A data base was constructed to store sample and mapping data. Forty additional samples of calcite and opal have been dated by the uranium-series method and yield finite ages between 50 ka and 430 ka. (Note: Two samples were older than 500 ka, which is the limit of the method.) Most of these samples were obtained from the outermost (youngest) parts of the specimens. Sixteen additional carbon-14 analyses were determined by accelerator mass spectrometry on calcite samples. These analyses yielded calculated ages between 25 ka and 52 ka, the latter being beyond the limit of the method. Two blank samples of old calcite were analyzed, and these yielded calculated ages greater than 50 ka and 51 ka. Most of the samples analyzed for carbon-14 also were obtained from the outermost (youngest) parts of crystals.

An additional dating technique was tested on calcite and opal deposits. The uranium contents of opal from the ESF are sufficiently large (up to approximately 300 ppm) that, in principal, the uranium-lead (U-Pb) dating technique could be used to determine ages of deposition if laboratory blanks are exceedingly low. Accordingly, collaborative work was started with the Jack Satterly Geochronology Laboratory at the Royal Ontario Museum in Toronto, Canada, to investigate the suitability of these opal deposits for U-Pb dating. Early results obtained on opal samples from the ESF were sufficiently encouraging to proceed with this dating technique, and 37 U-Pb ages now have been obtained for samples from the northern part of the ESF main drift, the northern bend, and the northwestern part of the north ramp. The dated subsamples were distributed approximately equally between outermost occurrences and occurrences embedded within the deposits. Ages obtained range from 100 ka to 9 Ma. The utility of this dating method is threefold: (1) for the younger samples, the U-Pb ages provide independent checks on the U-series ages, (2) the U-Pb ages provide further constraints and testing of the continuous-growth model described in Paces et al. (1996a), and (3) reliable ages of the embedded opal allow the calibration of the depositional stratigraphy of these deposits. Temporal calibration of the depositional stratigraphy of the calcite and opal occurrences is essential for calculating accurate rates of accumulation that can be inverted to flux estimates, and for establishing a time framework for the stable and radiogenic isotopic records that are isolated in the calcite deposits. These latter records relate to climatic conditions that existed at the surface of Yucca Mountain when the parent water of the calcite deposits was introduced by infiltration.

As reported in Paces et al. (1996a), initial uranium-234/uranium-238 ratios of the dated calcite and opal samples vary systematically as a function of stratigraphic position and depth in the ESF. Samples from the Tiva Canyon Tuff, the PTn and the upper part of the Topopah Spring Tuff have uranium-234/uranium-238 ratios generally between 1.5 and 2.0, which are similar to the values of surficial calcite. Samples from the repository horizon have larger initial uranium-234/uranium-238 ratios as great as 9.3. Samples analyzed this fiscal year indicate that this depth-stratigraphic variation will be repeated in the southern part of the ESF main drift and in the south ramp with uranium-234/uranium-238 ratios decreasing through the PTn and the Tiva Canyon Tuff. A provisional conceptual model has been constructed to explain the systematic variation of initial uranium-234/uranium-238 ratios in the calcite and opal deposits. Water infiltrating at the surface acquires the uranium-234/uranium-238 values of surficial materials (calcrete and bedrock coatings) that typically have ratios between 1.4 and 1.8. As small volumes of water percolate downward through the rock mass, uranium-234/uranium-238 ratios will gradually evolve to larger values as a result of alpha recoil and progressive uptake of uranium-234 from decay-damaged sites on pathway surfaces (the bulk rock has uranium-234/uranium-238 ratios of unity reflecting secular equilibrium of the system). Large volumes of percolating water would strip available uranium-234 resulting in uranium-234/uranium-238 ratios similar to those at or near the surface. Therefore, the large uranium-234/uranium-238 values are indicative of relatively small volumes of percolating water. In addition, abundant lithophysal cavities in the crystal-poor upper lithophysal zone of the Topopah Spring Tuff (tptpul) may effectively slow downward percolation, allowing fracture water the opportunity to spread laterally and interact with the upward-mitigating gas phase

To further understand the relationship between matrix water and fracture water in the unsaturated zone, pore-water salts were extracted from borehole USW SD-7 core from the Tiva Canyon Tuff, the PTn, the Topopah Spring Tuff, and the CHn for strontium-isotope analyses. These soluble salts form when the core dries and the pore water evaporates. The salts are redissolved using high-purity deionized water, and the strontium is then separated from the other cations using ion exchange methods. A systematic but variable increase in delta strontium-87 ($\delta^{87}Sr$) downward from the top of the hole through the crystal poor lower lithophysal zone (Tptpll) of the Topopah Spring Tuff offers important insight into the role of the PTn in percolation through the repository block. Pore water in the upper part of Tiva Canyon Tuff has a $\delta^{87}Sr$ value of +3.5, which increases to +3.8 near the bottom of the unit. Through the PTn, the $\delta^{87}Sr$ values increase monotonically from +3.8 at the top to +4.5 at the bottom. Downward through the crystal-poor middle nonlithophysal zone (Tptpmn) and crystal-poor lower lithophysal zone (Tptpll) of the Topopah Spring Tuff, $\delta^{87}Sr$ values are mostly in the range of +4.5 to +4.8. $\delta^{87}Sr$ then decreases in the basal vitrophyre of the Topopah Spring Tuff and the upper part of the CHn, but the scatter increases substantially. A provisional interpretation suggests that systematic isotopic variation above and below the PTn is caused by two distinct isotopic signatures of infiltrating water. One component infiltrates largely through bedrock surfaces and acquires $\delta^{87}Sr$ values of calcite bedrock coatings (+3.5) whereas the other component infiltrates the thick calcitic soils of alluvial valley fill acquiring a $\delta^{87}Sr$ of +4.5. The systematic variation in $\delta^{87}Sr$

through the PTn can be interpreted as an indication of storage, mixing, and lateral flow that may take place in this unit.

Carbon, oxygen, and strontium isotopic data continue to be collected for calcite fracture- and cavity-filling samples to characterize the changing character of the downward percolating fluids as a function of time. As discussed in Paces et al. (1996a), correlations among $\delta^{13}C$, $\delta^{18}O$, and $\delta^{87}Sr$ data indicate that the variations in isotopic content of the calcite samples do indeed relate to conditions that existed at the surface when the parent waters infiltrated. Using uranium-lead and uranium-series ages to estimate ages of subsamples of calcite for several occurrences reveals a positive correlation between $\delta^{13}C$ and age in the range of 25 ka to 9 Ma (heavier carbon with increasing age). Only the oldest subsample (9 Ma) has a $\delta^{13}C$ value outside the range of possible soil carbon. The other samples record a long-term change in climate and vegetation with a suggestion of a more abrupt change at approximately 3 Ma. The somewhat larger values of $\delta^{13}C$ prior to 3 Ma are compatible with a climate that would support grasslands. This preliminary attempt to use "age calibrated" microstratigraphy in the calcite and opal deposits to establish a correlation of subsurface conditions with surface climate is encouraging.

Forecast: The Ash Meadows area south of Highway 95 appears to be a modern analog of the Stateline area deposit, and will be reexamined to test the interpretation of the field relationships, geochronology, geochemistry, and paleontology of the Stateline sites. Geochronologic analysis will continue of samples from the deposits near Stateline, along the north side of Highway 95 near Lathrop Wells, and from Crater Flat. These records of deposition timing at these sites will be used in conjunction with textural and field observations, ostracode counts, and stable isotope studies to correlate past climates with the chemistry and depositional environments of these localities. Of particular importance will be the collection of strontium-isotope data from the stratigraphic sections near Stateline to determine the relative contributions of the Amargosa-Oasis Valley and Fortymile Wash-Yucca Mountain ground-water systems during deposition of the Stateline sequences.

Samples from the ESF south ramp and from the Northern and Southern Ghost Dance Fault Alcoves will continue to be collected and analyzed. Samples from the south ramp will be used to advance an understanding of the role of the PTn in controlling percolation through the Topopah Spring Tuff. The Northern and Southern Ghost Dance Fault Alcoves will allow direct access to and sampling of the Ghost Dance fault underground. Line surveys continue in the south ramp and the adequacy of the survey data for the main drift will be evaluated. If necessary, the 30-m intervals currently being surveyed every 100 m will be extended to 60-m intervals or full coverage to improve the abundance statistics. Uranium-series, carbon-14, and uranium-lead dating of calcite and opal will continue for samples in the south ramp and in the Northern and Southern Ghost Dance Fault Alcoves. The U-Pb dating will emphasize the determination of accurate ages for opal lenses embedded in calcite-dominated deposits so that the carbon, oxygen, and strontium isotopic records contained therein can be placed in a timeframe for correlation with surficial climate records. Work will continue on the development and application of the thorium-230/radium-226 system to opal and calcite to better constrain the continuous-deposition model. Continued emphasis will be placed on refining a conceptual model(s) of percolation using the ages and distribution of the low-temperature mineral occurrences. Efforts will be made to construct scenarios for different climate regimes and to develop more quantitative numerical models to simulate the temporal and spatial distribution of percolation flux.

3.4.8 Study 8.3.1.5.2.2 - Characterization of the Future Regional Hydrology Due to Climate Changes

The objective of this study is to characterize impacts of potential future climate changes on site unsaturated zone hydrology and regional and site surface-water system and saturated zone hydrology.

Activity 8.3.1.5.2.2.1 - Analysis of future surface-water hydrology due to climate changes. The objectives of this activity are to simulate past changes in runoff and surface-water storage (lakes) resulting from past climatic change; and to use the relationship between paleoclimate and paleosurface-water conditions to predict the impact of future climatic conditions on the surface-water hydrology at the site.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.5.2.2.2 - Analysis of future unsaturated zone hydrology due to climate changes. The objective statement for this activity has been deleted.

As reported in Revision 9 of the Site Characterization Program Baseline (DOE, 1995a), this activity has been deleted; the scope of work will be performed under Study 8.3.1.2.2.9 (see Section 3.1.13 of this progress report). See also Appendix H for the Site Characterization Program Baseline history.

Activity 8.3.1.5.2.2.3 - Evaluation of possible future changes of the climate and regional geologic framework on the regional saturated zone hydrology. The objectives of this activity are to reconstruct paleohydrologic conditions at Yucca Mountain and use these conditions together with the paleoclimatic conditions reconstructed as a basis to predict the impact of future climatic conditions on the saturated zone hydrologic system; to synthesize the existing paleohydrologic data through the use of numerical simulation techniques to determine effects that greater recharge would have on water-table altitude, ground-water flow paths, and hydraulic gradients between Yucca Mountain and the accessible environment; and to evaluate possible regional tectonic and thermal events that may produce prolonged or transient effects on the regional water level.

Progress is reported under Activity 8.3.1.2.1.4.4 (Regional three-dimensional hydrologic modeling) in Section 3.1.4 of this progress report.

Forecast: No activity is planned for this study during FY 1997.

3.5 EROSION (SCP SECTION 8.3.1.6)

The changes to the Erosion Program since the SCP was issued are summarized in Appendix A, Section A.1.5.

The scope of work envisioned for the erosion studies has been completed. As a result of the study plan work scope consolidation effort, no study plans were developed in this program. During the previous reporting period, the NRC provided the DOE with an Issue Resolution Status Report that described the basis for the NRC determination that adequate information existed to close all the open items associated with this issue (NRC, 1996d). [See Progress Report #14 (DOE, 1996g)].

Forecast: No further work is planned for the Erosion Program.

3.6 POSTCLOSURE TECTONICS (SCP SECTION 8.3.1.8)

Changes to the Postclosure Tectonics Program since the SCP was issued are summarized in Appendix A, Section A.1.7.

3.6.1 Study 8.3.1.8.1.1 - Probability of Magmatic Disruption of the Repository

The objective of this study is to assess the probability of future magmatic activity with respect to siting of a potential repository for the storage of high-level radioactive waste at Yucca Mountain.

Activity 8.3.1.8.1.1.1 - Location and timing of volcanic events. The objective of this activity is to synthesize the data collected by other activities on the dating, location, and volume of late Cenozoic volcanic events in the region surrounding the site.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.1.1.2 - Evaluation of the structural controls of basaltic volcanic activity. The objective of this activity is to investigate the time-space patterns of past volcanic activity in the Yucca Mountain region and the possible structural controls of volcanic centers and potential future centers at and adjacent to Yucca Mountain.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.1.1.3 - Presence of magma bodies in the vicinity of the site. The objective of this activity is to review geophysical and geochemical data collected in the vicinity of the site to assess whether there are any indications of the presence of crustal magma bodies that could be the source of future volcanic activity.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.1.1.4 - Volcanism probability studies. The objective of this activity is to revise the estimates of the probability of volcanic disruption of a repository site at Yucca Mountain, incorporating newly acquired data on the age, location, and volume of volcanic centers in the Nevada Test Site region and the results from activities investigating the possibility of structural controls of sites of volcanic activity and the presence of magma bodies in the Yucca Mountain area. These data may result in modifications of the area ratio and the rate of volcanic activity used in the probability formula.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No further work is planned for this study.

3.6.2 Study 8.3.1.8.1.2 - Physical Processes of Magmatism and Effects on the Repository

The objective of this study is to gather data on the potential effects of magmatic activity on the proposed repository. The data will be used to assess the consequences of such an eruption on repository performance.

This study was terminated at the end of FY 1996.

Activity 8.3.1.8.1.2.1 - Eruptive effects. The objective of this activity is to determine the effects of hydrovolcanic, Hawaiian, Strombolian, and violent Strombolian eruptions of basaltic magma on a repository. The results will be available for use in performance assessment calculations of possible radiological releases.

Activity 8.3.1.8.1.2.2 - Subsurface effects of magmatic activity. The objective of this activity is to evaluate the subsurface effects of emplacement of basalt dikes and intrusive bodies through and adjacent to a potential repository. This study will assess the mechanisms of incorporating waste in magma, the geometry of basalt intrusions, and hydrothermal effects on waste isolation of basalt intrusions through or near a repository.

Activity 8.3.1.8.1.2.3 - Magma system dynamics. The objectives of this activity are to evaluate the dynamics of basaltic magmatism, including tracing the processes of formation of basalt magma through generation in the mantle, ascent through the mantle and crust, potential storage in the mantle and crust, and eruption at the earth's surface.

Forecast: No further work is expected for this study.

3.6.3 Study 8.3.1.8.2.1 - Tectonic Effects: Evaluations of Changes in the Natural and Engineered Barrier Systems Resulting from Tectonic Processes and Events

The objective of this study is to assess the probability and effects of tectonic processes and events that could result in adverse effects on waste package lifetime, average percolation flux rate over the repository, altitude of the water table, local fracture permeability, effective porosity, and rock geochemical properties.

Activity-level progress narratives for this study have been combined into a single discussion because of the close interrelation of the phenomena being considered under credible tectonic scenarios that could affect hydrologic properties and characteristics.

Work continued to advance the analysis of possible credible tectonic scenarios described by G. E. Barr et al. (1996) that could affect hydrologic properties and characteristics, including changes in water-table elevation and accumulation of perched water. Most activity was devoted to assimilating prior results and reviewing ongoing work and interpretations pertinent to scenario development and evaluation. Two major issues were considered: (1) likelihood of the magnitude and recurrence of Quaternary tectonic phenomena, only three of which are significant: basaltic volcanism, local fault displacement and associated fracturing and block tilting, and ground motion or dynamic-stress effects caused by earthquakes; and (2) linkage between seismic activity and volcanism.

The issues of fault displacement and ground motion presently are being assessed through the probabilistic seismic hazard analysis of Yucca Mountain (see Study 8.3.1.17.3.6 in Section 3.13.8 of this progress report). Large faulting events at Yucca Mountain are likely linked to volcanism, but they are related by way of a common crustal-extension mechanism, not through direct cause and effect. Therefore, there is probably a threshold mechanism in effect: below a certain strain threshold faulting occurs without volcanism, and it is probably not distributed and probably involves segmented fault activity (O'Leary, 1996).

With respect to possible effects of tectonic processes and events on the hydrologic system, there are several open issues (G. E. Barr et al., 1996). Among these are perched water, the large hydraulic gradient in the saturated zone, and fault control on flow in the saturated zone. Even though perched water has not as yet been evaluated exhaustively in scenarios, it is important because a tectonic event could alter the connection between perched water and either its source or its drain. Such changes could lead to a diversion of the source pathways or a reduction in the effectiveness of the drain, both of which could result in perched water occurring close to the repository where it could interact with repository heat or participate in the flow pathways responsible for transport. Both the large hydraulic gradient and fault control on saturated zone flow are open issues because of uncertainty about their durability in response to seismic activity.

Activity 8.3.1.8.2.1.1 - Analysis of waste package rupture due to tectonic processes and events. The objective of this activity is to collect and synthesize data that can be used to assess the probability and effects of tectonic processes and events that could result in adverse impacts on waste package lifetime and performance.

See discussion under the heading for Study 8.3.1.8.2.1.

Activity 8.3.1.8.2.1.2 - Analysis of the effects of tectonic processes and events on average percolation flux rates over the repository. The objective of this activity is to analyze and assess the probability and effects of tectonic initiating events that may result in changes in the average percolation flux rate at the top of the Topopah Spring welded hydrogeologic unit.

See discussion under the heading for Study 8.3.1.8.2.1.

Activity 8.3.1.8.2.1.3 - Analysis of the effect of tectonic processes and events on water-table elevation. The objective of this activity is to produce analyses and assessme . of the probability that tectonic initiating events could result in significant changes in the elevation of the water table, changes in the hydraulic gradient, the creation of discharge points in the controlled area, or the creation of perched aquifers in the controlled area.

See discussion under the heading for Study 8.3.1.8.2.1.

Activity 8.3.1.8.2.1.4 - Analysis of the effects of tectonic processes and events on fracture permeability and effective porosity. The objective of this activity is to address possible changes in fracture permeability and effective porosity caused by tectonic events and processes.

See discussion under the heading for Study 8.3.1.8.2.1.

Activity 8.3.1.8.2.1.5 - Analysis of the effects of tectonic processes and events on rock geochemical properties. The objective of this activity is to provide assessments of the initiating events related to local changes in distribution coefficients resulting from tectonic processes and events.

See discussion under the heading for Study 8.3.1.8.2.1.

Forecast: Work will continue on evaluation of tectonic-scenario logic trees with respect to recent hydrologic modeling results, data acquired from ESF investigations, tectonic models evaluation, and probabalistic seismic hazard analysis results. Scenario evaluation will include describing the processes and ranges of possible consequences under two environments: the ambient environment (present day deformation rates and climate) and a projected anthropogenic environment (present day tectonic and climate conditions plus the full thermal load). A third evaluation factor involves a wet-climate environment.

3.6.4 Study 8.3.1.8.3.1 - Analysis of the Effects of Tectonic Processes and Events on Average Percolation Flux Rates Over the Repository

The objective of this study is to produce analysis and assessments of the probability and effects of tectonic initiating events that may result in changes in the average percolation flux rate at the top of TSw2.

This study has been combined with Study 8.3.1.8.2.1 (Section 3.6.3 of this progress report).

Forecast: No further work is planned under Study 8.3.1.8.3.1.

3.6.5 Study 8.3.1.8.3.2 - Analysis of the Effects of Tectonic Processes and Events on Changes in Water-Table Elevation

The objective of this activity is to provide analyses and assessments of the probability that tectonic initiating events could result in significant changes in the elevation of the water table, changes in the hydraulic gradient, the creation of discharge points in the controlled area, or the creation of perched aquifers in the controlled area.

This study has been combined with Study 8.3.1.8.2.1 (Section 3.6.3 of this progress report).

Forecast: No further work is planned under Study 8.3.1.8.3.2.

3.6.6 Study 8.3.1.8.3.3 - Analysis of the Effects of Tectonic Processes and Events on Local Fracture Permeability and Effective Porosity

The objective of this study is to address possible changes in fractures, or rock mass permeability and effective porosity caused by tectonic events and processes.

This study has been combined with Study 8.3.1.8.2.1 (Section 3.6.3 of this progress report).

Forecast: No further work is planned under Study 8.3.1.8.3.3.

3.6.7 Study 8.3.1.8.4.1 - Analysis of the Effects of Tectonic Processes and Events on Rock Geochemical Properties

The objective of this study is to provide assessments of the initiating events related to local changes in distribution coefficients resulting from tectonic processes and events.

This study has been combined with Study 8.3.1.8.2.1 (Section 3.6.3 of this progress report).

Forecast: No further work is planned under Study 8.3.1.8.4.1.

3.6.8 Study 8.3.1.8.5.1 - Characterization of Volcanic Features

The objective of this study is to provide data on the age, location, eruptive history, and volume of young volcanic rocks in the vicinity of the site. These data will be used to refine the calculations on the probability of igneous or volcanic events occurring in the controlled area and penetrating the repository.

Activity 8.3.1.8.5.1.1 - Volcanism drillholes. The objective of this activity is to investigate the origin of four or five aeromagnetic anomalies found in Crater Flat and the Amargosa Valley. Data from this work will be used to refine probability calculations, to refine geophysical models of the Yucca Mountain region, to evaluate the tectonic setting of volcanic centers, and to test concepts of the temporal and geochemical evolution of basalts in the Yucca Mountain region.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.5.1.2 - Geochronology studies. The objective of this activity is to establish the chronology of basaltic volcanism and the youngest silicic volcanic activity in the Yucca Mountain region. These data will be used to revise the recurrence rate of the volcanic probability calculations and to determine the age of cessation of silicic volcanic activity. Further studies are required for three topics: (a) the age of the Quaternary volcanic events in the Yucca Mountain region, (b) the age and eruption chronology of the youngest (<0.5 Ma) volcanic event in the Yucca Mountain area; and (c) the age of the youngest silicic volcanic activity in the region with emphasis on the Black Mountain caldera or young silicic rocks that may be encountered in shallow volcanic drillholes.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.5.1.3 - Field geologic studies. The objective of this activity is to establish the field geologic relations and the eruptive history of basaltic volcanic centers in the Yucca Mountain region.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.5.1.4 - Geochemistry of scoria sequences. The objective of this activity is to determine the geochemistry of scoria sequences of different ages at the Lathrop Wells center and older centers in the Crater Flat volcanic zone. The models will be used to test geologic assumptions made for the probability calculations and the time-space tectonic model for the distribution of basaltic volcanism. In addition, the data on the geochemistry of the scoria sequences also will be used to correlate basaltic ash in fault trenches with their correct eruptive source.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.8.5.1.5 - Geochemical cycles of basaltic volcanic fields. The objective of this activity is to determine the time-space geochemical variations of the volcanic fields of the southern Great Basin.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Further work on this study has been deferred indefinitely; no work is planned for FY 1997.

3.6.9 Study 8.3.1.8.5.2 - Characterization of Igneous Intrusive Features

The objective of this study is to gather data concerning the presence of thermal anomalies in the area and data on the geochemical and physical effects of intrusions on the surrounding rock. The evidence for the presence or absence of thermal anomalies will be used as part of the evaluation of the presence of significant magma bodies in the area and their relation to the probability of future volcanic events.

No progress was made during the reporting period; this was an unfunded study.

Forecast: No activity is planned for FY 1997.

3.6.10 Study 8.3.1.8.5.3 - Investigation of Folds in Miocene and Younger Rocks of Region

The objective of this study is to establish the regional pattern and rate of Neogene folding.

This study contains one activity that relies on available data; no unique data are to be acquired. Therefore, no study plan will be developed.

Activity 8.3.1.8.5.3.1 - Evaluation of folds in Neogene rocks of the region. The objective of this activity is to establish the pattern, rate, amplitude, and wavelength of post-middle Miocene folding in the region.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No further work is expected in this study.

3.7 HUMAN INTERFERENCE (SCP SECTION 8.3.1.9)

Changes to the Human Interference Program since the SCP was issued are summarized in Appendix A, Section A.1.8.

The studies in this program are intended to assess the survivability of a long-term surface marker system, the natural resource potential, and the impacts of potential human activities in the foreseeable future that could affect the waste isolation capabilities of the site. The natural resource assessment includes metallic resources, water resources, industrial rocks and minerals, and energy resources (such as, oil, gas, coal, tar sands, geothermal energy, and uranium). Studies on the inadvertent human interference issue will evaluate the feasibility of a surface marker system that would warn future generations of a hazard and assess the potential for inadvertent human interference associated with exploration for or exploitation of natural resources.

3.7.1 Study 8.3.1.9.1.1 - An Evaluation of Natural Processes that Could Affect the Long-Term Survivability of the Surface Marker System at Yucca Mountain

The objective of this study is to provide information on the currently or potentially active natural processes at Yucca Mountain capable of adversely affecting the long-term survivability of the surface marker system. This study will synthesize data obtained from other activities to be undertaken in support of several investigations. Suitable locations of the monuments for the surface marker system will be determined.

Activity 8.3.1.9.1.1.1 - Synthesis of tectonic, seismic, and volcanic hazards data from other site characterization activities. The objective of this activity is to identify the potential locations of faulting and volcanic eruption or intrusion that could occur where they could affect the marker system.

Locations for the surface markers were recommended using information in Fehr et al. (1996). The recommended locations for the markers are on bedrock, at higher elevations, and spaced such that adjacent markers are visible from each location.

Activity 8.3.1.9.1.1.2 - Synthesis evaluation of the effects of future erosion and deposition on the survivability of the marker system at Yucca Mountain. The objective of this activity is to determine the effects of future erosion and deposition on the topographic elements of the controlled area boundary at Yucca Mountain. The available information is being evaluated to identify the optimum locations for the markers.

Suggested locations for the surface markers were indicated on the map included in the feasibility report (Fehr et al., 1996).

Forecast: Additional data evaluation and probability studies will be completed as needed to support design of the permanent marker system

3.7.2 Study 8.3.1.9.2.1 - Natural Resource Assessment of Yucca Mountain, Nye County, Nevada

The objective of this study is to identify and assess the natural resource potential at the proposed repository site at Yucca Mountain. The assessment will examine the present and foreseeable future resource potential of the controlled area in comparison with the surrounding area and the general geologic setting. The information and data obtained in this study will provide the basis for probabilistic calculations for evaluating the potential for inadvertent human interference.

The geochemical survey for the metallic resources study and the oil and gas (petroleum) resources study, as well as a water resources study, are in progress. The natural resources synthesis report discussed in the forecast, will present the results of all the component studies.

Activity 8.3.1.9.2.1.1 - Geochemical assessment of Yucca Mountain in relation to the potential for mineralization. The objective of this activity is to conduct a geochemical sampling program to evaluate the potential for precious, base, and strategic metals; energy resources; and industrial mineral resources in the vicinity of Yucca Mountain. Specific objectives include (a) selecting a suite of elements for analysis in a geochemical sampling program on the basis of known commodities that occur in silicic tuffs and/or trace elements indicative of commodities that occur in the tuffs. (b) developing a field program to include a systematic and biased sampling of surface materials, (c) generating a first-order geochemical data base for selected elements obtained from surface and subsurface sampling within the vicinity of Yucca Mountain, (d) evaluating the data base in conjunction with geological and geophysical data obtained from other site characterization activities to determine if additional data are needed for an evaluation of natural resources, and (e) evaluating the potential for the occurrence of natural resources in the vicinity of Yucca Mountain based on an analysis of the geochemical data.

Laboratory geochemical analyses were completed as part of the metallic resources activity (Activity 8.3.1.9.2.1.5), and a data base compiling all analyses conducted for the project was constructed. These results will be reported in the metallic and mined energy resources report. A summary of this report will be incorporated into the natural resources synthesis report, which is discussed in the forecast.

Activity 8.3.1.9.2.1.2 - Geophysical/geological appraisal of the site relative to mineral resources. The objective of this activity is to qualitatively evaluate the available geophysical data base as it relates to Study Plan 8.3.1.9.2.1. Geologic models derived from geophysical data will be evaluated for their impact on mineral resources.

The results of the geologic and geophysical appraisal of the site and comparison of the site geology with the regional geology will be reported in the metallic and mined energy resources report and summarized in the natural resources synthesis report, which is discussed in the forecast.

Activity 8.3.1.9.2.1.3 - Assessment of the potential for geothermal energy at Yucca Mountain, Nevada. The objective of this activity is to evaluate regional ambient heat flow and local heat flow anomalies. This activity assesses the geothermal regime in terms of its energy resource potential for either hydrothermal or conductive reservoir thermal systems.

This activity was reported as completed in Progress Report #14 (DOE, 1996g, p. 3-111). The findings of this report will be incorporated into the natural resources synthesis report, which is discussed in the forecast.

Activity 8.3.1.9.2.1.4 - Assessment of hydrocarbon resources at and near the site. The objectives of this activity are to determine the potential for the presence or absence of suitable source rocks, reservoir rocks, and traps and seals at or near the site; to determine the potential for occurrence of conventional hydrocarbon resources (crude oil and natural gas) at and near the site; and to provide necessary data for the overall mineral and energy resource assessment to be performed.

Information is being compiled on the hydrocarbon potential of Yucca Mountain and the potential compared with other areas having hydrocarbon reserves is in progress. A report is being prepared that will synthesize data on the hydrocarbon resources at the site and assess the oil and gas potential of the site. The potential for undiscovered hydrocarbon resources in the controlled area will be estimated by comparing the site geology and geochemistry with deposit models and with areas in the region that contain known deposits in similar geologic settings. A summary of this report will be incorporated into the natural resources synthesis report, which is discussed in the forecast.

Activity 8.3.1.9.2.1.5 - Mineral and energy assessment of the site, comparison to known mineralized areas, and the potential for undiscovered resources and future exploration. The objective of this activity is to integrate the data and information collected from the geochemical assessment, geophysical/geologic assessment, geothermal energy assessment, hydrocarbon assessment, and the water resources assessment (Study 8.3.1.9.2.2).

Work during the period focused on reviewing drafts of reports from the above activities and compiling data for the natural resources synthesis report, discussed in the forecast.

Forecast: The natural resources synthesis report will be developed as a final assessment of the potential for natural resources, using site-specific data compared with regional geologic data and models of resource deposit genesis. Assessments of the metallic resources, oil and gas resource potential, mined energy resources, and water resources of the site will be combined with previously completed assessments of industrial rocks and minerals and geothermal resources.

This synthesis report will integrate geological, geophysical, and mineralogical information from the literature with data from the site, including new geological and geochemical data from Yucca Mountain and areas of known mineralization in the region. The potential for undiscovered deposits in the controlled area will be evaluated by comparing its geology and geochemistry with mineral deposit models and with areas in the region that contain known metal deposits in similar geologic settings. The completion of the natural resources synthesis report will conclude the st: ies planned in the SCP for natural resources. The report is scheduled to be completed in June, 1997 during the next reporting period.

3.7.3 Study 8.3.1.9.2.2 - Water Resource Assessment of Yucca Mountain, Nevada

The objective of this study is to use available data to estimate the future supply, demand, and value of the ground-water resource proximal to Yucca Mountain.

Activity 8.3.1.9.2.2.1 - Projected trends in local and regional ground-water development and estimated withdrawal rates in southern Nevada, proximal to Yucca Mountain. The objectives of this activity are to assess the current and projected supply and demand situation in the foreseeable future for ground water in the geohydrologic study area and to estimate the value of the ground-water resource.

A report is being prepared that describes ground water supply and demand at Yucca Mountain and in the surrounding region, including the Las Vegas Valley. Estimated future demand, value, and probable locations and rates of future exploitation will be included. These findings of the ground-water resources final report will be summarized in the natural resources synthesis report (see forecast for Section 3.7.2 of this progress report).

Forecast: The report on the ground-water resources will be completed. Information from this report will be summarized in the natural resources synthesis report (see Section 3.7.2 of this progress report).

3.7.4 Study 8.3.1.9.3.1 - Evaluation of Data Needed to Support an Assessment of the Likelihood of Future Inadvertent Human Intrusion at Yucca Mountain as a Result of Exploration and/or Extraction of Natural Resources

The objective of this study is to compile and analyze data to assess the likelihood of inadvertent human interference in the Yucca Mountain vicinity.

Because no unique data were to be acquired by this study, no study plan was developed. The entire scope of the study was transferred to Study 8.3.1.9.2 (see Appendix A.1.8.2).

Activity 8.3.1.9.3.1.1 - Compilation of data to support the assessment calculation of the potential for inadvertent human intrusion at Yucca Mountain. The objectives of this activity are to determine the maximum drilling density and frequency (drillholes per square kilometer per 10,000 years) that can be reasonably assumed for a repository at Yucca Mountain; and to determine the extent to which future ground-water withdrawals will modify the expected ground-water flow paths.

The scope of this activity was transferred to Study 8.3.1.9.2.1

Forecast: No further work is planned for this study (see Appendix A.1.8.2).

3.7.5 Study 8.3.1.9.3.2 - An Evaluation of the Potential Effects of Exploration for, or Extraction of, Natural Resources on the Hydrologic Characteristics at Yucca Mountain

The objective of this study is to assess, in qualitative or quantitative terms, the effects of exploiting natural resources known or believed to be present at Yucca Mountain. Consideration of the effects of resource exploitation or extraction are limited to changes in the hydrologic, geochemical, and rock characteristics.

No unique data are to be acquired by this study; thus, no study plan was developed. The entire scope of this study was transferred to Study 8.3.1.9.2.1 (see Appendix A.1.8.2).

Activity 8.3.1.9.3.2.1 - An analysis of the potential effects of future ground-water withdrawals on the hydrologic system in the vicinity of Yucca Mountain. The objective of this activity is to determine the potential effects of future ground-water withdrawals on the hydrologic system at Yucca Mountain. Effects of the withdrawals will be defined qualitatively and quantitatively.

Work conducted during the reporting period in support of this activity is described under Study 8.3.1.9.2.1 (see Appendix A).

Activity 8.3.1.9.3.2.2 - Assessment of initiating events related to human interference that are considered not to be sufficiently credible or significant to warrant further investigation. The objective of this activity is to demonstrate that those initiating events that have been identified (Table 8.3.1.9-1, SCP) for the human interference issue are not considered sufficiently credible or significant to necessitate additional investigation.

Work conducted during the reporting period in support of this activity is described under Study 8.3.1.9.2.1 (see Appendix A).

Forecast: No work is planned for FY 1997. The inputs to computer modeling for varying water withdrawal assumptions and input bounds will be supplied under Study 8.3.1.9.2.2 and will be reported in the ground-water resources final report, scheduled for next reporting period.

3.8 METEOROLOGY (SCP SECTION 8.3.1.12)

Changes to the Meteorology Program since the SCP was issued are summarized in Appendix A, Section A.1.9.

Four studies and one investigation in the Meteorology Program (8.3.1.12) include work to describe current local and regional meteorological conditions. Three of the studies (8.3.1.12.1.1, 8.3.1.12.1.2, and 8.3.1.12.4.1) and the investigation (8.3.1.12.3) are controlled by the Scientific Investigation Implementation Package for Regional Meteorology (CRWMS M&O, 1995a) and the Nevada Work Instruction: "Acquisition and Analysis of Regional Meteorological Data." The fourth study, Meteorological Data Collection at the Yucca Mountain Site (8.3.1.12.2.1), is controlled by the corresponding study plan.

3.8.1 Study 8.3.1.12.1.1 - Characterization of the Regional Meteorological Conditions

The major objective of this study is to collect and analyze meteorological data from various locations surrounding Yucca Mountain and to characterize the regional meteorology needed for site characterization performance assessment and design. This characterization will provide a regional overview of wind flow patterns and other parameters (primarily related to atmospheric dispersion and surface facility design) associated with meteorological conditions at Yucca Mountain. For additional descriptions of the objectives, refer to the Scientific Investigation Implementation Package for Regional Meteorology (CRWMS M&O, 1995a).

The climatological analyses and preparation of the Engineering Design Climatology and Regional Meteorological Conditions report continued. Select data identified in the meteorological data synthesis report (CRWMS M&O, 1996j) are being included in the climatological analyses. The analyses include re-evaluating 1985 to 1992 data from the meteorological stations operated by the Project Environmental Field Programs Division.

Forecast: The climatological analyses and reporting phases of this study will be completed.

3.8.2 Study 8.3.1.12.1.2 - Plan for Synthesis of Yucca Mountain Site Characterization Project Meteorological Monitoring

The objective of this study is to develop a plan that provides for the coordination of meteorological monitoring efforts by various Project participants during site characterization.

The work in this study was combined with other studies in the Scientific Investigation Implementation Package for Regional Meteorology (CRWMS M&O, 1995a) (see Section 3.8 of this progress report).

Forecast: Future work on this study will be controlled through Study 8.3.1.12.1.1 (see Section 3.8.1 of this progress report).

3.8.3 Study 8.3.1.12.2.1 - Meteorological Data Collection at the Yucca Mountain Site

The primary objective of this study is to provide site-specific data to resolve design and performance issues associated with preclosure radiological safety, including estimating potential radiological dosage related to repository operations. Data from this study are also used to respond to air quality permit requirements of the State of Nevada covering site characterization activities.

Activity 8.3.1.12.2.1.1 - Site meteorological monitoring program. The objective of this activity is to collect meteorological data at potential locations of surface facilities and at sufficient additional locations to characterize the meteorological conditions, including wind flow patterns and atmospheric dispersion characteristics, in the vicinity of Yucca Mountain.

The meteorological monitoring program continued at the nine sites that have been active since 1993. Data were also supplied to the design and environmental groups. Field operations for 17 recording precipitation gauge stations previously operated by the USGS as part of Study 8.3.1.2.1.1 were added to the site meteorological monitoring activities at the beginning of 1997. Data from these stations will be made available to the investigators working in unsaturated zone infiltration studies. The data recording at the original 9 full meteorological stations was modified to include the specific data records requested by the USGS.

Select meteorological data from the monitoring network were included in the two quarterly ambient air monitoring reports submitted to the State of Nevada during the reporting period. These reports fulfill requirements of the State Air (ality Permit (No. AP9611-0573) required for continuing site-disturbing activities. No significant increase in inhalable particulate matter (PM_{10}) concentration was noted during the reporting period.

Activity 8.3.1.12.2.1.2 - Data summary for input to dose assessments. The objective of this activity is to process the collected meteorological data into a format and content that will be useful in assessing radiological impacts, as required by design and performance issues.

Data requests were fulfilled for data from the site network formatted to be compatible with atmospheric dispersion models being evaluated for impact assessment purposes.

Forecast: Data collection and reporting will continue at the 9 monitoring stations and the additional 17 recording precipitation stations. The data formatting activity will be performed once the airborne radiological impact assessment method has been determined.

3.8.4 Investigation 8.3.1.12.3 - Studies to Provide Data on the Location of Population Centers Relative to Wind Patterns in the General Region of the Site

The objective of this investigation is to provide data on wind flow patterns in the general region of Yucca Mountain. These patterns are needed to identify areas that could be impacted by airborne radiological material released from surface or underground facilities at Yucca Mountain.

The work in this study was combined with other studies in the Scientific Investigation Implementation Package for Regional Meteorology (CRWMS M&O, 1995a) (see Section 3.8 of this progress report).

Forecast: Future work on this study will be controlled and tracked through Study 8.3.1.12.1.1 (see Section 3.8.1 of this progress report).

3.8.5 Study 8.3.1.12.4.1 - Characterize the Potential Extreme Weather Phenomena and Their Recurrence Intervals

The objective of this study is to evaluate existing historical meteorological and climatological records, technical publications, and other relevant information to quantify the extreme weather phenomena that may be expected at the Yucca Mountain site and to determine their recurrence intervals.

This study analyzes existing meteorological data and technical publications to estimate the potential for occurrences and types of extreme weather events that might affect repository operations. The work in this study was combined with other studies in the Scientific Investigation Implementation Package for Regional Meteorology (CRWMS M&O, 1995a) (see Section 3.8 of this progress report).

Forecast: Future work on this study will be controlled and tracked through Study 8.3.1.12.1.1 (see Section 3.8.1 of this progress report).

3.9 OFFSITE INSTALLATIONS AND OPERATIONS (SCP SECTION 8.3.1.13)

Changes to the Offsite Installations and Operations Program since the SCP was issued are summarized in Appendix A, Section A.1.10.

The objective of this program is to provide information required to support resolution of design and performance issues related to offsite radiological safety, including (a) evaluations of offsite accident initiators, their probabilities and potential impacts; (b) assessments of routine releases from nuclear operations; (c) assessments of the onsite impacts of nonrepository-related routine and potential accidental releases of radioactive material; and (d) collection of agricultural and cultural data to support the calculation of the dose to the public from releases at the Yucca Mountain Site.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Activities are planned to begin in the last half of FY 1997 for completion in FY 1998 to meet objectives described in the SCP. See Appendix A, Section A.1.10

3.10 SURFACE CHARACTERISTICS (SCP SECTION 8.3.1.14)

Changes to the Surface Characteristics Program since the SCP was issued are summarized in Appendix A, Section A.1.11.

3.10.1 Study 8.3.1.14.2.1 - Exploration Program

The objective of this study is to conduct an exploration program for the characterization of the soil and rock conditions that will influence or be influenced by the construction of the surface facilities and the subsurface facilities. The exploration program study will consist of site reconnaissance and preliminary and detailed exploration

Activity 8.3.1.14.2.1.1 - Site reconnaissance. The objective of this activity is to review existing site information and conduct field reconnaissance to establish a preliminary exploration program to include further topographic and geologic mapping, subsurface drilling, test pits, trenching, and geophysical methods.

The reconnaissance activities identified in the objective statement have been completed. No new activity occurred during the reporting period.

Activity 8.3.1.14.2.1.2 - Preliminary and detailed exploration. The objective of this activity is to obtain sufficient surface and subsurface data to prepare a preliminary design for the ESF surface and subsurface access facilities. Preliminary designs based on these explorations will be suitable for economic and technical feasibility reports and Project planning reports.

No progress was made during the reporting period because no activity was planned for FY 1997.

Forecast: No additional activity is planned for this study in FY 1997 although future work may be required to support design of waste handling facilities in FY 1999.

3.10.2 Study 8.3.1.14.2.2 - Laboratory Tests and Material Property Measurements

The objective of this study is to conduct laboratory tests and material property measurements on representative samples of soil and rock. These tests and measurements are intended to determine physical, mechanical, and dynamic properties. Additional tests and measurements will be conducted on soils to determine index properties and moisture-density compaction curves for potential fill material.

Activity 8.3.1.14.2.2.1 - Physical property and index laboratory tests. The objective of this activity is to measure the soil or rock weight and volume components using physical property tests.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.14.2.2.2 - Mechanical and dynamic laboratory property tests. The objective of this activity is to measure in the laboratory the static and dynamic deformation and strength characteristics of soil and rock samples obtained from the exploratory program. The results of this testing will be used to evaluate bearing capacity, earth pressures, shear strength parameters, slope stability, settlement and swelling potentials, and the dynamic characteristics of the soil and rock.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No additional activity is planned for FY 1997 although future work may be required to support design of waste handling facilities in FY 1999.

3.10.3 Study 8.3.1.14.2.3 - Field Tests and Characterization Measurements

The objective of this study is to conduct field tests and characterization measurements. These field tests are intended to determine the in situ physical, mechanical, and dynamic properties of the soil and rock.

Activity 8.3.1.14.2.3.1 - Physical property field tests and characterization measurements. The objectives of this activity are to classify and describe the soil and rock conditions in the field and to determine their physical properties. The results of these tests and measurements will be used to estimate the engineering characteristics of the soil and rock. In addition, these properties and measurements will aid in the grouping of soil and rock into stratigraphic units and the extrapolation of results from a restricted number of mechanical and dynamic properties tests to zones of soil and rock with similar material properties.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.14.2.3.2 - Mechanical property field tests. The objective of this activity is to measure the deformation and strength characteristics of in situ soil and rock conditions. The results of this testing will be used to design ESF surface facilities and underground openings.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.14.2.3.3 - Geophysical field measurements. The objectives of this activity are to obtain measurements of the compressional and shear wave velocities and to determine the velocity structure in the area of the ESF surface facilities and subsurface ramps and shafts. These methods may also be used to profile the alluvium-bedrock contact, to locate discontinuities or other structural features, and to determine the depth, thickness, and lateral extent of soil and rock stratigraphic units.

This activity has been completed. The compilation and analysis of geophysical data are described under Activity 8.3.1.4.1.2 (Integration of Geophysical Activities), discussed in Section 3.3.2 of this progress report.

Forecast: No additional activity is planned for this study in FY 1997 although future work may be required to support design of waste handling facilities in FY 1999.

3.11 THERMAL AND MECHANICAL ROCK PROPERTIES (SCP SECTION 8.3.1.15)

Changes to the Thermal and Mechanical Rock Properties Program since the SCP was issued are summarized in Appendix A, Section A.1.12.

3.11.1 Study 8.3.1.15.1.1 - Laboratory Thermal Properties

The objective of this study is to provide laboratory characterization of thermal conductivity and heat capacity and provide data to describe the spatial variability of these parameters. To accomplish this, porosity, grain density, and the heat capacity and thermal conductivity of zero-porosity material must also be characterized.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpmn, Tptpln) are found in Buesch et al. (1996a).

Activity 8.3.1.15.1.1.1 - Density and porosity characterization. The objective of this activity is to obtain data on density and porosity and to evaluate the spatial variability thereof. Data will contribute to determining in situ thermal properties (porosity and grain density), vertical in situ stress (bulk density), and radiation-shielding properties (bulk density).

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.15.1.1.2 - Volumetric heat capacity characterization. The objective of this activity is to obtain data for volumetric heat capacity and to evaluate the spatial variability thereof. The data will be used in calculations of the thermal response to the presence of heat-producing waste in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln, see Geologic/Lithologic Stratigraphy, MO 9510R1B 002.004).

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.15.1.1.3 - Thermal conductivity characterization. The objective of this activity is to obtain data for thermal conductivity and to evaluate the spatial variability thereof. The data will be used in calculations of the thermal response to the presence of heat-producing waste in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln).

Equipment to measure thermal conductivities of geologic specimens has been set up and is being calibrated.

Forecast: Testing is planned for rocks from the Thermal Testing Facility, from the Southern Ghost Dance Fault Alcove, and for rocks from lithostratigraphic units below the repository level sampled in USW SD-7. Characterization of grain densities for units below the repository level are planned for samples recovered from USW SD-7. The data from this study will be used in calculations of thermal stress and deformation associated with the temperature field produced by the presence of heat-producing waste in unit TSw2 (lithostratigraphic units

Tptpmn, Tptpll, and Tptpln) (see Geologic/Lithologic Stratigraphy, MO 9510R1B00002.004). Samples from the Thermal Testing Facility and the Southern Ghost Dance Fault Alcove of the main drift of the ESF will be tested to determine the lateral variability and anisotropy of thermal conductivity and thermal expansion properties. Thermal conductivity of test specimens from the drift-scale test area of the Thermal Testing Facility will be determined. These results will be reported under SCP Study 8.3.4.2.4.4 (Section 5.2.5 of this progress report).

3.11.2 Study 8.3.1.15.1.2 - Laboratory Thermal Expansion Testing

The objective of this study is to provide laboratory characterization of thermal expansion behavior and the spatial variability thereof. This information will be used to establish the testing frequency at the main test level in the ESF.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpmn, Tptpln) are found in Buesch et al. (1996a).

Activity 8.3.1.15.1.2.1 - Thermal expansion characterization. The objective of this activity is to obtain data for thermal-expansion behavior and to evaluate the spatial variability thereof. The data will be used in calculations of thermal stress and deformation associated with the temperature field produced by the presence of heat-producing waste in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln).

This activity has been incorporated into Study 8.3.1.15.1.1, discussed in Section 3.11.1 of this progress report.

3.11.3 Study 8.3.1.15.1.3 - Laboratory Determination of Mechanical Properties of Intact Rock

The objective of this study is to provide laboratory characterization of the mechanical properties of intact rock, including the spatial variability and the effects of changes in environmental conditions.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpmn, Tptpln) are found in Buesch et al. (1996a).

Activity 8.3.1.15.1.3.1 - Compressive mechanical properties of intact rock at baseline experiment conditions. The objective of this activity is to obtain data for the compressive mechanical properties of intact rock and the spatial variability thereof for baseline experiment conditions. These data will be used in mechanical and thermomechanical calculations of stresses and deformations induced by the presence of underground openings in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln) and overlying units and by the presence of heat-producing waste in unit TSw2.

No progress was made during the reporting period. Results from this activity will be reported under SCP Study 8.3.4.2.4.4 (Section 5.2.5 of this progress report).

Activity 8.3.1.15.1.3.2 - Effects of variable environmental conditions on mechanical properties. The objective of this activity is to evaluate the effects of varying sample size, strain rate, temperature, confining pressure, lithophysal content, saturation state, and anisotropy on compressive mechanical properties. Data will be used in mechanical and thermomechanical calculations of stresses and deformations induced by the presence of heat-producing waste in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln).

No progress was made during the reporting period; this was an unfunded activity.

Forecast: If funding is provided in the next fiscal year, the main drift of the ESF will be characterized and the spatial variability and effects of anisotropy, pressure, and temperature will be examined. Uniaxial compressive strength and elastic moduli measurements for test specimens from the drift-scale test area of the Thermal Testing Facility will be conducted during the second half of FY 1997.

3.11.4 Study 8.3.1.15.1.4 - Laboratory Determination of the Mechanical Properties of Fractures

The objective of this study is to provide laboratory characterization of the mechanical properties of fractures, including the spatial variability and the effects of changes in environmental conditions.

Symbols for the lithostratigraphic units of the Paintbrush Group exposed at Yucca Mountain (e.g., Tptpmn, Tptpln) are found in Buesch et al. (1996a).

Activity 8.3.1.15.1.4.1 - Mechanical properties of fractures at baseline experiment conditions. The objective of this activity is to obtain data for the mechanical properties of fractures, and the spatial variability thereof, for baseline experiment conditions. The data will be used in mechanical and thermomechanical calculations of the stresses and deformations induced by the presence of underground openings in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln) and overlying units and by the presence of heat-producing waste in unit TSw2.

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.15.1.4.2 - Effects of variable environmental conditions on mechanical properties of fractures. The objective of this activity is to evaluate the effects of varying normal stress, displacement rate, temperature, sample size, fracture roughness, and saturation state on the mechanical properties of artificial and natural fractures. The data will be used in mechanical and thermomechanical calculations of stresses and deformations induced by the presence of underground openings in unit TSw2 (lithostratigraphic units Tptpmn, Tptpll, and Tptpln) and overlying units and by the presence of heat-producing waste in unit TSw2.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: If funding is provided in the next fiscal year. the main drift of the ESF will be characterized to assess the lateral variability of mechanical properties of fractures at baseline conditions. Additional potential work includes determining flow properties through fractures as a function of fracture aperture.

3.11.5 Study 8.3.1.15.1.5 - Excavation Investigations

The objective of this study is to obtain site-specific information concerning the behavior of underground excavations in the proposed repository horizon and overlying units. Most of the data will be used for testing of computer codes that will be used to predict mechanical behavior of the rock mass. In addition, some of the information will directly demonstrate performance of repository-scale openings.

Activity 8.3.1.15.1.5.1 - Access convergence experiment. The objective of this activity is to monitor rock-mass deformation around accesses as they are constructed. The intent in the SCP (DOE, 1988) was to measure rock stress changes and deformation of the rock mass as a shaft was progressively constructed to the repository horizon. Since the SCP, changes in the ESF design and testing strategy have lead to a new ESF design where shafts are replaced by a single tunnel. This change made most of this activity unnecessary or a duplication of the efforts being pursued under Study 8.3.1.15.1.8 (Section 3.11.8 of this progress report).

No progress was made during the reporting period; this was an unfunded activity.

Activity 8.3.1.15.1.5.2 - Demonstration breakout room. The objective is to demonstrate constructability and stability of underground rooms with cross-sectional dimensions equivalent to those of a repository in both lithophysae-rich and lithophysae-poor material.

This test was based on data needs related to the ESF design described in the SCP (DOE, 1988) and has been deferred because access to the ESF is by ramps rather than shafts. Test information related to this activity is being collected under Study 8.3.1.15.1.8 (see Section 3.11.8 of this progress report).

Activity 8.3.1.15.1.5.3 - Sequential drift mining. The objectives of this activity are to obtain data on the deformation response of drifts with cross-sectional dimensions equivalent to those of a repository in welded tuff, to use the data in model evaluation activities, and to demonstrate constructability and stability of repository-sized drifts in lithophysae-poor material. Data will contribute to validating computer models to be used to calculate mechanical responses, as well as empirical evaluations related to nonradiological health and safety.

Detailed planning for this test was completed in FY 1996. The test plan is described in the Thermal Test Design and Layout Report (CRWMS M&O, 1996k). In November 1996, three multipoint borehole extensometers were installed in 27-m-long boreholes extending from the access-observation drift to within 1 m of the rib of the heated drift (when excavated). Installation occurred before the heated drift was excavated. The access-observation drift, heated drift, and cross drift are parts of the Thermal Testing Facility that will be used in the upcoming drift-scale thermal test discussed in Section 3.11.6, Activity 8.3.1.15.1.6.5. The principal part of the heated drift was excavated between December 1996 and February 1997. Data were recorded as the excavation passed each multipoint borehole extensometer gauge station. These data provide a baseline for rock movement resulting from the excavation of the drift that can be compared with subsequent movement resulting from heating the drift. As excavation proceeded, tape extensometer stations were established inside the heated drift. These measurements of cross-drift closure were monitored as excavation continued. With the exception of the tape extensometer stations, the instrumentation installed for this test will remain in place and also be used for the planned drift-scale thermal test to measure rock response to repository emplacement room heating.

Forecast: Data will continue to be collected from installed instrumentation through the testing period of the heated drift.

3.11.6 Study 8.3.1.15.1.6 - In Situ Thermomechanical Properties

The objective of this study is to obtain data on in situ thermal and thermomechanical properties for units TSw1 and TSw2. Properties to be obtained include heat capacity, thermal conductivity, and thermal expansion. Additional heater experiments will be conducted to characterize the waste container environment.

Planning, design, and equipment installation continue for testing of the heated drift. During the reporting period, detailed equipment procurement and installation schedules were developed.

Activity 8.3.1.15.1.6.1 - Heater experiment in unit TSw1. The objectives of this activity are to estimate the in situ thermomechanical properties of lithophysae-rich tuff (unit TSw1) and to evaluate the thermal and mechanical response of this tuff unit to elevated temperatures.

No progress during the reporting period because of the rapid progress of ESF construction.

Activity 8.3.1.15.1.6.2 - Canister-scale heater experiment. The objective of this activity is to obtain thermal and thermomechanical rock-mass measurements of the effects of thermal inputs on a representative scale in lithophysae-poor tuff (unit TSw2). The data will be used to evaluate thermal and thermomechanical properties of the rock mass and to evaluate the thermal and thermomechanical models.

In early FY 1997, final as-built locations for all the instrumentation and the surveyed geometry of the test block were determined (SNL, 1996a). An initial set of pre-test analyses was performed before the test installation was completed and the single-heater test was initiated (SNL, 1996b). Using the as-built configuration, a second set of pretest analyses was performed (SNL, 1996c). These analyses predict the response of each of the approximately 350 thermal/mechanical sensors.

Approximately six months of data have been recorded from the test thus far. Most sensors appear to be operating properly. Temperatures recorded on the heater package surface and in the surrounding rock were within bounds established by pre-test predictions. As reported in Progress Report #15 (DOE, 1997e), however, the geochemical sensors have not operated correctly since installation. Data through the end of November 1996 has been reduced and reported (SNL, 1997). Data acquired through the end of February 1997 is being reduced and prepared for submittal to the records system. Comparisons of analysis predictions and data collected through the end of November 1996 are under way.

Data reduced and analyzed thus far indicate that the single-heater test is working as designed. Preliminary results indicate that (a) convective heating effects are smaller than predicted from the thermohydrologic-hydrologic analyses, possibly a result of vapor escaping the block or through fracture systems not accounted for in the modeling, (b) at temperatures above boiling, temperature predictions agree quite well with measured values, (c) as expected, rock-mass thermal expansion coefficient appears to be somewhat (10 to 20 percent) less than the thermal expansion coefficient of the intact rock, (d) rock-mass modulus, as measured by a borehole jack, does not yet show any significant effect of temperature, and (e) the rock bolts installed in the heated zone of the test show a greater reduction in tension load than bolts installed on the ambient rock (because the expansion of steel is greater than the rock in this temperature range).

Activity 8.3.1.15.1.6.3 - Yucca Mountain heated block. The objective of this activity is to estimate in situ mechanical and thermomechanical properties of unit TSw2 and to test thermomechanical models.

The data needs and objectives related to this test have been combined into the ESF tests, under Activities 8.3.1.15.1.6.2 and 8.3.1.15.1.6.5. (Note: The status of Thermal Testing Facility design and construction are described in Sections 7.1.2 and 7.3.3, respectively, of this progress report.)

Activity 8.3.1.15.1.6.4 - Thermal stress measurements. The objective of this activity is to monitor thermally induced stress in jointed welded tuffs in an accelerated test.

The data needs and objectives related to these measurements have been incorporated into those of the ESF thermal tests conducted under Activities 8.5.1.13.1.6.2 and 8.3.1.15.1.6.5.

Activity 8.3.1.15.1.6.5 - Heated room experiment. The objectives of this activity are (a) to evaluate the thermomechanical response of welded tuff around repository openings to expected repository conditions during both construction and operation; (b) to develop a data base for evaluating thermal and thermomechanical design analyses and methods applicable for repository considerations; and (c) to use actual site data in predicting drift response and support/rock interactions during construction, operation, retrievability, and postclosure.

Data that will provide a baseline for rock movement were acquired under Activity 8.3.1.15.1.5.3 (see Section 3.11.5 of this progress report) as the drifts for the drift-scale test were excavated. These data are being analyzed.

Final layouts for the instrument arrays have been completed and are being documented. Specifications for each instrument and for the canister and wing heaters have been completed and are being procured. The installation of the first multipoint borehole extensometer parallel to the drift in a 47-m-long borehole began.

Pre-test analyses of the heated drift are under way using site-specific rock property data and planned geometry and gauge locations. These analyses will be completed and documented in June 1997.

Forecast: The single-heater test will continue, and data will be analyzed and compared with pre-test modeling results. Additional modeling will be performed using site-specific data for these comparisons. A design for the drift-scale test will be completed and documented. The instrument boreholes will be drilled and instruments installed in the heated drift.

3.11.7 Study 8.3.1.15.1.7 - In Situ Mechanical Properties

The objective of this study is to obtain in situ measurements of the mechanical properties of the rock mass for unit TSw2.

Activity 8.3.1.15.1.7.1 - Plate loading tests. The objective of this activity is to measure the deformation modulus of the rock mass and to evaluate the zone of increased fracturing adjacent to underground openings. A plate loading test will be conducted as part of the heated drift test. This will allow the measurement of rock-mass modulus under both ambient and heated conditions. The planning for this test is taking place as part of the consolidated thermal test.

The consolidated thermal test consists of two tests: the single-heater test (Activity 8.3.1.15.1.6.2), and the drift-scale test (Activity 8.3.1.15.1.6.5) as described in the In Situ Thermal Testing Program Strategy (DOE, 1995c). A preliminary design of the plate loading test associated with the ESF thermal test was developed and documented in the Thermal Test Design and Layout Report (CRWMS M&O, 1996k).

The niche for the plate loading test in the Thermal Testing Facility has been constructed. Instrumentation is being specified for future procurement, and the reaction frame is being designed. The test will not be conducted until the heaters in the heated drift have been operating for approximately one year so that the rock mass activated by the test will be at elevated temperature.

Activity 8.3.1.15.1.7.2 - Rock-mass strength experiment. The objective of this activity is to evaluate the mechanical behavior of the rock mass or its components by using experiments to obtain information related to the mechanical strength of single joints and to multiply jointed volumes of rock.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: The plate loading test design will be finalized. The test equipment will be partially installed in late FY 1997.

3.11.8 Study 8.3.1.15.1.8 - In Situ Design Verification

The objectives of this study are (a) to investigate the effects of the spatial variability of the rock on drift stability, mining activities, and ground supports; (b) to evaluate techniques for underground excavation and ground support, for selecting ground supports to be used in different rock types, and for monitoring drift stability; (c) to quantify the emanation of radon into repository drifts and observe its dispersion with airflow; and (d) to measure parameters needed to design repository ventilation systems.

Geotechnical design verification activities have been conducted in the ESF to provide data that can be used to confirm adequacy of design, construction, and long-term performance since the beginning of ESF construction. The data from these activities will also be used to support repository design and to validate the ESF design.

Activity 8.3.1.15.1.8.1 - Evaluation of mining methods. The objective of this activity is to develop a recommendation for mining in the repository by monitoring and evaluating mining activities in the ESF and by conducting mining investigations.

Evaluations of rock mass quality have been keeping pace with ESF excavation. These evaluations were needed for correlation with other studies and to substantiate ground support decisions made by the ESF constructor. Preliminary data were submitted daily and the reviewed data were submitted monthly to the records center.

The blast seismic monitoring and blast damage assessment were performed during construction of the Northern Ghost Dance Fault Alcove and parts of the Thermal Testing Facility, including the access-observation drift, connecting drift, and heated drift. Blast monitoring consists of (a) measuring the near-field and far-field peak particle velocities caused by drill and blast excavation methods, (b) visually inspecting nearby boreholes, and (c) using spectral analysis of the surface wave to estimate blast damage behind the rock surface. Peak particle velocities were measured, and a scaled distance model was developed. Measured velocities have shown considerable variability, and no work to determine the distribution of the particle velocities has been done. Peak particle velocities serve as empirical indicators of the extent of blast damage in the surrounding rock. Blast monitoring in the access-observation drift of the Thermal Testing Facility and Northern Ghost Dance Fault Alcove indicated that damage was generally below the rock damage criteria at 1 m into the rock around the alcove opening. The far-field results obtained on the full-face rounds did not indicate blast anomalies that would result in high peak particle velocity values. Preliminary data were submitted on a daily basis, and the reviewed data were submitted as a Technical Data Information Form package to the Records Processing Center

Activity 8.3.1.15.1.8.2 - Monitoring of ground-support systems. The objective of this activity is to develop recommendations for a ground-support methodology to be used in repository drifts, based on evaluations of the ground-support methodology being used in the ESF and experimentation with other ground-support configurations. Support systems will be recommended, as will methods of selecting appropriate supports for the ground conditions encountered.

Project scientists continued to monitor installed geotechnical instruments and installed new geotechnical instruments closely following the ESF excavation. The geotechnical instrumentation included convergence and strain gauges on steel sets installed as part of the ground support, instrumented rock bolts, and rock bolt load cells. Strain gauges were installed on selected steel sets either before or after installation. For pre-installed strain gauges, installation loads were measured before, during, and after installation. Readings from these instruments have identified no significant concerns regarding the integrity of ground support.

In situ stress measurements were performed in the Thermal Testing Facility at the end of the access-observation drift. The measurements were performed using a mini-fracture method. The results were consistent with previous estimates of horizontal in situ stress (maximum horizontal stress being approximately 50 percent of the vertical overburden stress) (see Section 8.3.1.15 of the SCP).

Activity 8.3.1.15.1.8.3 - Monitoring drift stability. The objectives of this activity are (a) to provide confidence in predictions of usability of the repository underground facilities for their 100-year operational life, (b) to contribute to evaluations of the effectiveness of mining methods and ground supports, (c) to calibrate and refine criteria for determining stability of the openings, and (d) to develop techniques for monitoring stability of the repository drifts.

Project scientists continued to monitor installed geotechnical instruments and installed new geotechnical instruments closely following the ESF excavation. The geotechnical instrumentation included multipoint and single-point extensometers and cross-drift convergence pins. Readings from these instruments have identified no significant concerns related to tunnel stability. Reviewed data were submitted quarterly to the Records Processing Center, with an annual summary report submitted at the end of FY 1996.

Activity 8.3.1.15.1.8.4 - Air quality and ventilation experiment. The objectives of this activity are to measure the rate of radon emanation from the repository host rock and to evaluate parameters and variables needed as input to or for testing of the models to be used to design the ventilation systems in the repository underground facility.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Monitoring of instrumentation in the ESF south ramp will continue until the tunnel boring machine emerges at the south portal. After that, monitoring of installed instrumentation will continue on a quarterly or semiannual basis.

3.11.9 Study 8.3.1.15.2.1 - Characterization of the Site Ambient Stress Conditions

The objective of this study is to characterize the ambient (pre-repository) state of stress of the Yucca Mountain host rock and surrounding units for use as initial conditions for geomechanical models used in the design and performance assessment of the repository underground facilities.

Activity 8.3.1.15.2.1.1 - Anelastic strain recovery experiments in core holes. The objective of these experiments using samples from core holes is to determine the horizontal stresses at Yucca Mountain and especially the spatial variability of these stresses. In situ stress data will contribute to the definition of initial and boundary conditions for mechanical and thermomechanical analyses.

This activity was developed to support the design and operation of the shaft configuration of the ESF and the potential repository Abandonment of the shaft configuration has eliminated the need for this activity.

Activity 8.3.1.15.2.1.2 - Overcore stress experiments in the Exploratory Studies Facility. The objectives of this activity are to determine the in situ state of stress above, within, and below the repository host rock in that part of the repository block penetrated by the ESF, and to evaluate the extent to which the ambient stress conditions are redistributed adjacent to excavations. In situ stress data will contribute to the definition of initial and boundary conditions for mechanical and thermomechanical analyses.

The scope of work for this activity has been transferred to Study 8.3.1.15.1.8 (see Section 3.11.8 of this progress report).

Forecast: No further work is planned for this study.

3.11.10 Study 8.3.1.15.2.2 - Characterization of the Site Ambient Thermal Conditions

The objective of this study is to evaluate available thermal data to determine the ambient (pre-repository) temperature and thermal conductivity of the Yucca Mountain host rock and surrounding units for use as initial conditions for thermomechanical models used in the design and performance assessment of the repository underground facilities.

Activity 8.3.1.15.2.2.1 - Surface-based evaluation of ambient thermal conditions. The objectives of this activity are to measure the spatial variation of temperature with depth in existing wells; provide baseline temperatures within the repository host rock and surrounding units; measure thermal conductivity (near 25°C) of core samples to check on independent thermal property determinations at various temperatures; and determine heat flow at Yucca Mountain.

Data collection for this activity has been transferred to Study 8.3.1.8.5.2. and pertinent activities are reported under that study (see Section 3.6.9 of this progress report).

Forecast: No additional activity is planned for this study.

3.11.11 Related International Thermal and Mechanical Rock Properties Work

As of November 8, 1995, the subsidiary agreements under which the cooperative work had been conducted were terminated, and all international collaboration was discontinued.

The OCRWM had bilateral agreements with Canada (Atomic Energy of Canada Limited [AECL]), Switzerland (Swiss National Cooperative for the Storage of Radioactive Waste [Nagra]), and Sweden (Swedish Nuclear Fuel and Waste Management Company [SKB]) and has participated in activities of international organizations such as the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA), the European Commission (EC), and the International Atomic Energy Agency (IAEA).

Forecast: No work is planned under the international program.

3.12 PRECLOSURE HYDROLOGY (SCP SECTION 8.3.1.16)

Changes to the Preclosure Hydrology Program since the SCP was issued are summarized in Appendix A, Section A.1.13.

3.12.1 Study 8.3.1.16.1.1 - Characterization of Flood Potential of the Yucca Mountain Site

The objective of this study is to evaluate the potential for flooding in the many small, dry, desert washes that drain Yucca Mountain. This evaluation will be used for designing the surface facilities for the proposed repository. Proper design for flood potential is necessary to ensure the safety of workers and surface facilities.

No activities have been funded under this study since FY 1993. Flood potential for the site was described and evaluated in the Technical Basis Report for Surface Characteristics, Preclosure Hydrology, and Erosion (DOE, 1995d).

Forecast: No activities are planned for this study.

3.12.2 Study 8.3.1.16.2.1 - Location of Adequate Water Supply for Construction, Operation, Closure, and Decommissioning of a Mined Geologic Disposal System at Yucca Mountain, Nevada

The objective of this study is to identify water supply sources for a potential repository. Four activities were identified in the SCP: (1) assess the cost, feasibility, and adequacy of using wells UE-25 J#12 and UE-25 J#13 as an alternative water supply; (2) identify a primary water source; (3) identify another alternative water source (other than wells UE-25 J#12 and UE-25 J#13); and (4) identify and evaluate the potential effects of repository-related withdrawals on the ground water flow system.

This study uses data from other studies; therefore no study plan will be developed. No activities have been funded under this study since FY 1991. Water resource data were reviewed in the Technical Basis Report for Surface Characteristics, Preclosure Hydrology, and Erosion (DOE, 1995d).

Forecast: No activities are planned for this study.

3.12.3 Study 8.3.1.16.3.1 - Determination of the Preclosure Hydrologic Conditions of the Unsaturated Zone at Yucca Mountain, Nevada

The objective of this study is to compile the data collected under Geohydrology Investigation 8.3.1.2.2 for input to Design Issue 4.4.

The purpose of this study was to describe ground-water conditions within and along the potential repository block. Work originally planned for this study has been incorporated into and performed under Study 8.3.1.2.2.9, discussed in Section 3.1.13 of this progress report.

Forecast: No further work is planned under this study.

3.13 PRECLOSURE TECTONICS (SCP SECTION 8.3.1.17)

Changes to the Preclosure Tectonics Program since the SCP was issued are summarized in Appendix A, Section A.1.14.

3.13.1 Study 8.3.1.17.1.1 - Potential for Ash Fall at the Site

The objective of this study is to provide required information on volcanic activity that could affect repository design performance.

No study plan will be developed for this activity. The work identified in this section has been completed on the basis of available data and was documented in Perry and Crowe (1987).

Activity 8.3.1.17.1.1.1 - Survey literature regarding Quaternary silicic volcanic centers in the western Great Basin. The objective of this activity is to compile information on Quaternary silicic volcanism in the western Great Basin, the reoccurrence of which might produce an ash fall at the site.

This activity has been completed (see Perry and Crowe, 1987).

Activity 3.1.17.1.1.2 - Assess potential ash-fall thickness at the site. The objective of this activity is ... duce an approximate probability-versus-thickness function for potential ash falls at the site and to estimate a particular ash-fall thickness that has less than one chance in ten of occurring in 100 years. These hazard estimates will be considered in the design of filters in the mining and surface-facility ventilation systems.

This activity has been completed (see Perry and Crowe, 1987).

Forecast: No further work is planned for this study.

3.13.2 Study 8.3.1.17.2.1 - Faulting Potential at the Repository

The objective of this study is to provide required information on fault displacement that could affect repository design or performance.

No study plan was developed. The work scope for this study has been combined with that of Study 8.3.1.17.3.6 (see Section 3.13.8 of this progress report).

Forecast: No further work is planned under this study.

3.13.3 Study 8.3.1.17.3.1 - Relevant Earthquake Sources

The objective of this study is to identify and characterize those earthquake sources relevant to seismic hazard analysis of the site (i.e., those sources that could cause significant surface fault displacement or ground shaking at the site)

The activities composing this study were completed in FY 1996. Results are documented in Whitney (1996). The results form part of the information base supporting the seismic hazard assessment for Yucca Mountain (see Section 3.13.8 of this progress report).

Activity 8.3.1.17.3.1.1 - Identify relevant earthquake sources. The objective of this activity is to identify earthquake sources that could generate significant surface fault displacements or severe ground motions at the site.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.3.1.2 - Characterize relevant earthquake sources. The objective of this activity is to characterize each relevant earthquake source identified in the previous activity by providing a spatial description (including an expected depth or depth range), an assessment of activity, evaluations of maximum earthquake magnitude, the size and location of expected coseismic displacements (for sources in or near the controlled area), and the recurrence rate for earthquakes associated with the source. The source characterization includes an evaluation of variability in and dependency of input parameters.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Forecast: No additional work is planned for this Study.

3.13.4 Study 8.3.1.17.3.2 - Underground Nuclear Explosion Sources

The objective of this study is to characterize the potential future underground nuclear explosions at the Nevada Test Site that would result in the most severe motions at the repository site.

The nuclear test-ban treaty signed by the President in September, 1996 precludes underground testing for the foreseeable future.

Forecast: No work is currently planned for this study.

3.13.5 Study 8.3.1.17.3.3 - Ground Motion From Regional Earthquakes and Underground Nuclear Explosions

The objective of this study is to select or develop ground-motion models that are appropriate for estimating ground motion at the site from earthquake and underground nuclear explosions. These models will be used to determine the relevancy of seismic sources to a deterministic seismic hazard analysis, identify controlling seismic events, constrain simulated ground motions from controlling seismic events, and estimate the probabilities of exceeding given ground-motion levels at the site.

The activities included in this study were completed in FY 1996. Results are documented in Whitney (1996) and Walck (1996). The results form part of the information base supporting the seismic hazard assessment for Yucca Mountain (see Section 3.13.8 of this progress report).

Activity 8.3.1.17.3.3.1 - Select or develop empirical models for earthquake ground motions. The objective of this activity is to select or develop empirical ground-motion models that are appropriate for estimating earthquake ground motion at the site. The models will predict ground motion as a function of earthquake magnitude and distance between the earthquake source and the site.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.3.3.2 - Select or develop empirical models for ground motion from underground nuclear explosions. The objective of this activity is to select or develop empirical ground-motion models that are appropriate for estimating ground motion at the site from underground nuclear explosions. The models will predict ground motion as a function of underground nuclear explosions yield and distance between the underground nuclear explosions and the site.

This work has been completed.

Forecast: No additional work is planned for this study.

3.13.6 Study 8.3.1.17.3.4 - Effects of Local Site Geology on Surface and Subsurface Motions

The objective of this study is to document systematic effects on surface and subsurface ground motions resulting from the local site geology.

The activities composing this study have been completed, and results are documented in Whitney (1996). The results form part of the information base supporting the seismic hazard assessment for Yucca Mountain (see Section 3.13.8 of this progress report).

Activity 8.3.1.17.3.4.1 - Determine site effects from ground-motion recordings. The objective of this activity is to determine, from ground-motion recordings, the systematic effects of local site geology on surface and subsurface motions and to identify any significant site-wide bias in ground-motion levels, as compared with average levels for the southern Great Basin.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.3.4.2 - Model site effects using the wave properties of the local geology. The objective of this activity is to develop a calibrated theoretical site-effects model for use in extrapolating the observations documented in Activity 8.3.1.17.3.4.1 to locations and depths where ground-motion predictions are needed, but where instrumental recordings are not available.

As part of the probabilistic seismic hazard analyses being conducted in Study 8.3.1.17.3.6 (Section 3.13.8 of this progress report), studies are in progress using resonant-column and dynamic-tensional shear testing to more accurately determine the effects of rock properties on attenuation of ground motion.

Forecast: No additional work is planned for this study.

3.13.7 Study 8.3.1.17.3.5 - Ground Motion at the Site From Controlling Seismic Events

The objective of this study is to identify the controlling seismic events and to characterize the resulting controlling ground motions. Controlling seismic events are those underground nuclear explosions or earthquakes that would generate the most severe ground motions at the site at frequencies of engineering significance.

Activity 8.3.1.17.3.5.1 - Identify controlling seismic events. The objective of this activity is to identify underground nuclear explosions or earthquakes that would produce the most severe ground motions at the site at frequencies of engineering significance. There may be more than one controlling seismic event because different events may generate the most severe ground motions in different frequency bands.

A working group was convened to select or develop a method for determining controlling earthquakes. Controlling earthquakes will be determined based on the results of Study 8.3.1.17.8.3.6 (see Section 3.13.8 of this progress report) as well as other available information.

Activity 8.3.1.17.3.5.2 - Characterize ground motion from the controlling seismic events. The objective of this activity is to generate suites of strong-motion time histories and corresponding response spectra representative in amplitude, frequency content, and duration of site ground motions that could be generated by the controlling seismic events.

The working group described in Activity 8.3.1.17.3.5.1 will also select or develop an appropriate method for determining seismic design inputs using the identified controlling earthquakes.

Forecast: Project staff will complete the development of the method to determine seismic design inputs and prepare a report describing the method. The method will be implemented in the first quarter of FY 1998.

3.13.8 Study 8.3.1.17.3.6 - Probabilistic Seismic Hazards Analysis

The objectives of this study are to quantify (a) the probability of experiencing ground motions of varying degrees of severity that might result from earthquakes of varying magnitudes and distances from the potential repository site, and (b) the potential for fault displacements of varying degrees of severity to disrupt the surface facilities or the underground repository. (Note that this study combines the objectives originally designated for Studies 8.3.1.17.3.6 and 8.3.1.17.2.1, Section 3.13.2 of this progress report).

Activity 8.3.1.17.3.6.1 - Evaluate ground-motion probabilities. The objectives of this activity are to (a) quantify the probabilistic vibratory ground motion values appropriate for seismic design of the potential repository structures, systems, and components, and (b) provide documentation of the bases for these determinations sufficient for regulatory review and licensing.

See discussion under Activity 8.3.1.17.3.6.2.

Activity 8.3.1.17.3.6.2 - Assessment of fault displacement hazard. The objective of this activity is to assess the fault displacement hazard for repository design.

The probabilistic seismic hazards assessment process involves using a panel of experts to develop interpretations and assessments of uncertainties required by the hazards calculations. One panel of experts has been characterizing seismic sources and fault displacements, and the other panel has dealt with vibratory ground motion. The development of interpretations is being facilitated through a series of structured workshops to evaluate available data, explore the range of interpretations allowed by the data, examine critically the interpretations proposed by the experts, and provide feedback on the implications of various interpretations for the seismic hazards at the site. The goal of this process is to have differences in interpretations of the experts be true differences in judgment and not differences in access to data, definitions, or understanding other experts' interpretations.

A historical earthquake catalogue for use in the probabilistic seismic hazards assessment was compiled from all available data sources including existing national and regional catalogues and special studies of individual earthquakes. The catalogue covered a region extending to 300 km radius from the site. A substantial effort was made to remove duplicate events and identify and delete all nuclear explosions. Aftershocks induced by the explosions were also removed using a space-time window. All events were converted to moment magnitude using a set of equations derived from existing literature and studies by the University of Nevada at Reno. The catalogue was declustered using two sets of algorithms. The resulting two catalogues were made available to the seismic source experts for their use in characterizing the background earthquake and areal source zones.

The probabilistic seismic hazards assessment for Yucca Mountain resumed. This assessment, which began in FY 1995 and was suspended in FY 1996 because of budget constraints, consists of two parts:

- Seismic source and fault displacement characterization
- Ground motion characterization.

For the seismic source and fault displacement characterization, six teams of three experts each were formed; team individuals provide a range of expertise (paleoseismology, regional geology and tectonics, seismology) needed to develop interpretations and evaluate uncertainties. For the ground-motion characterization, seven individual experts were selected.

During the reporting period, four workshops and a field trip were held:

1. Seismic Source Characterization Hazard Methodologies Workshop

2. Seismic Source Characterization Alternate Models and Interpretations Workshop and Associated Field Trip

3. Seismic Source Characterization Preliminary Interpretations Workshop

4. Methods, Models, and Preliminary Interpretations Workshop on Ground Motions at Yucca Mountain.

The workshop on hazard methodologies was held October 16-18, 1996, in Salt Lake City, Utah. This workshop identified and outlined methods and approaches to characterize seismic sources for ground-motion and fault-displacement hazards assessments.

The Seismic Source Characterization Preliminary Interpretations Workshop was held January 6-8, 1997, in Salt Lake City, Utah. This workshop (a) allowed the expert teams to present and discuss their preliminary interpretations regarding several key issues in seismic-source characterization; (b) trained the teams in the process of elicitation and uncertainty characterization; and (c) presented additional information and recent analyses important to Yucca Mountain seismic-hazards assessments. Presentations were made primarily by the expert panel members, followed by group discussions on promoting interaction and common understanding among the experts.

The Ground Motion Models and Interpretations Workshop was held January 9-10, 1997, in Salt Lake City, Utah. The purpose of this workshop was to identify and outline methods and approaches to characterize numerical and empirical models for assessing ground-motion attenuation, path effects, and site response for Yucca Mountain.

Formal elicitation of experts' interpretations regarding seismic-source characterization started January 21, 1997, at the offices of Geomatrix Consultants, San Francisco, California. The elicitations were a means to collect and assemble the input for calculations of ground-motion and fault-displacement hazards by development of preliminary logic trees characterizing interpretations and assessments of uncertainty. The elicitation meetings facilitated the development of a logic tree structure for each expert or expert team and initiated the process of including weighted interpretations in the tree. Formal elicitations continued into February, at which time preliminary calculations were performed and the results returned to the experts for sensitivity analyses and feedback discussions.

Proceedings of these workshops and the field trip are described in four reports (CRWMS M&O, 1996m and 1996n; CRWMS M&O, 1997b and 1997c).

Forecast: The experts' assessments will be finalized following feedback workshops on the seismic-source and ground-motion characterizations. The final assessments will be used as input to final calculations of the annual probabilities of varying levels of ground motion and fault displacement at Yucca Mountain. The seismic hazard assessment process and its results will be documented in a report that will describe the logic basis and data used by the experts (or expert teams) in developing their assessments. The formal elicitation process, evaluations, probabilities, and calculations of potential ground motions and fault displacements will be completed in the second half of FY 1997. The report on the probabilistic seismic hazards assessment will be prepared, technically reviewed, and submitted to YMSCO in August 1997.

Principal results of the probabilistic seismic hazard assessment also will be summarized in Progress Report #17.

3.13.9 Study 8.3.1.17.4.1 - Historical and Current Seismicity

The objective of this study is to compile information on recorded earthquakes near Yucca Mountain. This information will be used to help identify and characterize potentially relevant earthquake sources for the probabilistic hazard analysis; to develop regional earthquake ground-motion models; and to determine local geologic and depth-of-burial effects on ground motion at the site.

Activity 8.3.1.17.4.1.1 - Compile historical earthquake record. The objective of this activity is to compile a record of historical seismic events in the southern Great Basin or within 100 km of Yucca Mountain that will indicate whether each cataloged seismic event is thought to be a natural earthquake, induced earthquake, under ground nuclear explosion, cavity collapse, or blast. For potentially damaging earthquakes ($M_M \geq 5.5$) in the study region, available information will be compiled on ground-motion intensity, availability of strong-motion records, and extent and style of faulting.

Work on this activity was completed in FY 1996. Preparation of an expanded earthquake catalog to support the probabilistic seismic hazard is discussed in Section 3.13.8 of Progress Report #14 (DOE, 1996g).

Activity 8.3.1.17.4.1.2 - Monitor current seismicity. The objective of this activity is to provide empirical information on the frequency of earthquake occurrence in the southern Great Basin; the orientation, depth, and style of faulting; how seismic-wave amplitudes scale with magnitude and attenuate with distance in the region; and how ground motions vary with depth and with surface geology in the site area.

Seismic monitoring of the southern Great Basin in the vicinity of Yucca Mountain continued. At the end of the reporting period, the network consisted of 24 3-component stations employing digital acquisition systems ... the field. Negotiations are in progress with the National Park Service for approval to install instruments at three remaining sites. The network covers an area with an approximate 50 km radius around Yucca Mountain. More than 800 earthquakes have been located within or near the network during the past 6-month period. The largest event occurred near the northeast corner of the Nevada Test Site on January 17, 1997 and had a local magnitude (M_L) of 3.5.

In November 1996, a significant earthquake swarm occurred to the north of the network, near Saucer Mesa, on the south flank of the Kawich Range 60 km north of the Yucca Mountain site. The largest event in this sequence had a local magnitude (M_L) of 4.1; several other events had magnitudes larger than 3.0. Several hundred aftershocks were recorded at the closest network station. The data from this event will be an important contribution to the determination of site response to seismicity.

Von Seggern and Smith (1997) describe the first full year (FY 1996) of operation of the digital network. This report covers numerous aspects of the network operation and performance, as well as the seismicity for FY 1996. More than 2100 earthquakes were located within or near the network and incorporated into the FY 1996 catalog. Focal mechanisms for more than 100 events were determined (a significantly larger number than in previous years). The detection capability in the immediate vicinity of Yucca Mountain was shown to have a threshold of M_L = -0.3 or less. Since the start of the digital network recording, 16 earthquakes within approximately 10 km of Yucca Mountain have been documented. These earthquakes were described in detail, and they confirm the overall low seismicity rate at Yucca Mountain. Using the high-quality digital network data, a moment-magnitude relation for the Yucca Mountain region was defined, and the rupture plane of the smallest detectable earthquake was determined to be on the order of 3 to 5 m. The smallest M_L observed was -0.9, corresponding to a seismic moment of roughly 3×10^{16} dyne-cm.

Tomography Using Teleseismic P Waves

A study of the Yucca Mountain region using teleseismic tomography was completed (Biasi, 1996). This study used relative teleseismic delay times from 117 earthquakes to invert for crustal and upper mantle velocity structure in the vicinity of Yucca Mountain. Both a regional model and a localized, more detailed model were developed. Results for the regional model show 2 to 3 percent high velocities extending to a depth of 200 km or more beneath the Timber Mountain/Silent Canyon caldera structure and 1 to 3 percent low velocities to the northeast, east, and southeast of Timber Mountain. At shallower depths (greater than about 70 km), 1 to 2 percent high velocities are also imaged to the west-northwest of Timber Mountain. Using the more detailed model, 1 to 3 percent low velocities are shown for crustal depths beneath Little Skull and Skull mountains. These velocities are interpreted to result from a structurally controlled low velocity zone associated with the Rock Valley fault. Shallow high velocities beneath Yucca Mountain connect with the Timber Mountain structure. For the detailed model, at mantle depths, the pattern of low velocities beneath Rock Valley and high velocities beneath Timber Mountain is still present. Although partial melt in small fractions cannot be eliminated because of resolution limitations, there is no large low-velocity zone under Crater Flat or Yucca Mountain that would suggest a major source of magma.

Network Processing Software

Beginning October 1, 1996, the Project converted to the Joint Seismic Project Center Datascope data base and seismic software for the routine processing of seismic events. Parametric data are now routinely stored in the Datascope tables and accessed with Datascope applications. This conversion has improved the efficiency of work and the ability to retrieve and analyze data.

Three major upgrades in software were made in this reporting period. The first was to JSPC 3.2 software, which addresses many previous "bugs" and enhances data processing. The second upgrade was to the Solaris operating system, specifically Solaris 2.5.1, on Project Sun computers. The third upgrade was the installation of the Earthworm system from the University of Alaska. This system was installed in parallel with the current processing scheme. Picking and associating signals with events is near real time. This new system will produce automatic locations and rapid display of data. Access to external data from other networks is inherent in the system; this capability will be very useful should a large earthquake occur near Yucca Mountain.

The paging system hardware, part of the Earthworm system, was installed and tested. This hardware provides the capability to send text, such as earthquake location and magnitude, to the alpha-numeric pager, providing a much higher level of information than simple beeper signals. The hardware also expedites University of Nevada, Reno, Seismological Laboratory response to major earthquakes, as well as to system problems.

Foam-Rubber Modeling of Normal-Fault Earthquakes

Physical models of faulting, such as foam-rubber models (as distinct from numerical or mathematical models), obey static and dynamic mechanical laws, and thus can be used to gain insight into the physical processes involved. In this study, surface accelerations from normal faults were compared with those from strike-slip geometries. The data show that surface accelerations near the normal fault trace are systematically lower, by an average factor of about 0.10, compared with the accelerations at the side sensors, which represent strike-slip motion. These results suggest that kinematic modeling of ground motion using classic dislocation theory should apply a significant adjustment of the fault slip time function on the shallow part of the fault. The results indicate that estimates of accelerations for normal faults should be scaled down considerably from value bases on current regression curves or simulations.

Physical modeling was performed using a shallow weak layer to verify the physical basis for assuming a long rise time and a reduced high-frequency pulse for the slip on the shallow parts of faults. The results indicate that a 2-km deep, weak zone along strike-slip faults could indeed reduce the high-frequency energy radiated from shallow slip. This effect can best be represented by superimposing a small-amplitude, short rise time pulse at the onset of a much longer rise-time slip. A weak zone was modeled by inserting weak plastic layers a few inches thick into the foam rubber model. The pulse observed in the model for the 3-in. layer has been reduced by a factor of 0.4 compared with the average value for the case with no weak zone; but, because only one observation was available, this value is quite uncertain. For the 6-in. weak zone, the average pulse is reduced by a factor of 0.46. For the 8-in. case, the reduction factor is 0.11. For the 12-in. case, the reduction factor is 0.045. These results indicated that the thicker the weak layer, the more difficult it is for a short rise-time acceleration pulse to push through the weak layer to the surface. Thus, this is an approximate justification for reducing the high frequency radiation from shallower parts of strike-slip faults.

Precarious Rocks

The distribution of fallen, precarious, and semiprecarious rocks around the Little Skull Mountain earthquake of 1992 has been mapped in more detail. The distribution is consistent with a well-constrained source model of this earthquake. Observations suggest that an earthquake of the magnitude of Little Skull Mountain is statistically unexpected in the time period of the historical monitoring (about 100 years) and that no events much larger than this earthquake have occurred in the last few thousand years.

Additional surveying of precarious rocks was conducted in northern Nevada, especially in regard to normal-fault regimes. Field measurements of quasi-static toppling accelerations indicate that, for instance, the accelerations of the 1915 Pleasant Valley Earthquake did not exceed 0.25 g in spite of its magnitude of 7.7. This has important implications for future normal-fault style events in the Basin and Range.

Historical Seismicity and History of Network Operations

A draft report has been completed on the historical seismicity of the southern Great Basin for the inclusion in Project Integrated Safety Assessment (in prep.) Section 3.3 (Site and Regional Geologic Description).

The report generally describes the historical seismicity within 300 km of Yucca Mountain, with a focus on significant regional earthquakes, and provides a detailed description of the activity within 100 km. Also included is a comprehensive bibliography of seismological studies and seismicity reports that have been conducted or compiled that are relevant to the 300 km and 100 km regions. A history of seismic network operations and network coverage for the historical period in southern Nevada and eastern California is also included; seismic network operations and data analysis procedures are described in the context of the quality and completeness of the historical record.

Strong Motion Network

Data from the Yucca Mountain strong motion network were downloaded in January. All the stations are operating except for the station at the Field Operations Center. The instrument at the Field Operations Center is being repaired for the second time. A Nevada Work Instruction has been prepared and reviewed to cover operation of this network and the handling of data from it.

Tunnel Monitoring

A three-component strong-motion sensor was installed during October 1996 in the Thermal Testing Facility. The installation is configured with three Kinemetrics FBA-11 1g sensors and a REFTEK 16-bit portable recorder with 528 Megabyte disk drive. Time was established outside of the ESF before the unit was installed, and some clock drift is expected. With clock corrections, however, accurate time should be recoverable. This strong motion station was visited in January, and the data collected up to that time were recovered.

Site Effects

The data from several hundred earthquakes recorded on the regional digital network have been organized and prepared for the analysis of the attenuation and site effect within the network region. Because of the decreased detection threshold and the quality of the recordings, several thousand seismograms can be used in the inversion. The records have been instrument corrected and the horizontal records have been rotated into the radial and transverse components of ground motion. Body-wave spectra have been calculated for both P and S wave arrivals and for pre-event noise for all three components of ground motion at each station for each event. Spectra have been calculated from 1 to 50 Hz, and these results have been organized in a data base. All information about the source locations has also been compiled into the data base, and the organization of the data has been established for the inversion procedure. This first study is designed to determine a general Q(f) for the region and the frequency-dependent site effects at each site of the regional digital network for weak ground motion. Because of the nature of the data from preliminary analyses, Q(z) and azimuth-dependent site effects may also be resolved with further study. A companion study is under way to calculate the site effects at the Yucca Mountain strong ground motion stations, and portable instruments are currently running in trigger mode at these sites.

Activity 8.3.1.17.4.1.3 - Evaluate potential for induced seismicity at the site. The objective of this activity is to evaluate the potential for human activity to significantly perturb natural seismic hazard at the site by inducing seismicity at or near the site. To date, the human activities that have been identified as having a potential to induce seismicity in the site region are the impoundment of Lake Mead, the testing of nuclear devices at the Nevada Test Site, and excavation of the repository.

Seismic monitoring within the ESF and by stations located near Yucca Mountain is providing a data base that will be used to assess the potential for excavation-induced seismicity. This data collection is being performed under Activity 8.3.1.17.4.1.2.

Forecast: When permits can be obtained for three stations in Death Valley National Park, the digital monitoring network will be extended to its full 27 stations. Roughly 1000 events are expected to be located in the last half of FY 1997, and with the full network the locating capabilities will be significantly improved for the Amargosa Desert, Funeral Mountains, and Bullfrog Hills. Within the next six months, location notification capability is expected at near real time for earthquakes within the network using Earthworm software. The site effects will be determined at the sites of the permanent network and the strong-motion network and Q(f) will be determined within the network. The transition of the University of Nevada, Reno, Seismological Laboratory studies to the Management and Operating Contractor should be formally complete, allowing data to be gathered solely under approved quality assurance procedures. Monitoring will continue within the ESF and at borehole UE-25 UZ#16 for unusual seismic activity.

3.13.10 Study 8.3.1.17.4.2 - Location and Recency of Faulting Near the Prospective Surface Facilities

The objective of this study is to identify a site in Midway Valley sufficiently large for surface facilities in which significant Quaternary faults are absent.

Activity 8.3.1.17.4.2.1 - Identify appropriate trench locations in Midway Valley. The objective of this activity is to identify appropriate trench locations at proposed locations for repository surface facilities that are important to safety through detailed geologic mapping.

This activity was completed in FY 1996. See Progress Report #15.

Activity 8.3.1.17.4.2.2 - Conduct exploratory trenching in Midway Valley. The objectives of this activity are to investigate the possible occurrence of late Quaternary surface fault rupture in the vicinity of planned surface facility locations important to safety and to identify sites without evidence of significant late Quaternary faulting. This activity will provide input into the location and design of surface facilities important to safety, particularly those associated with waste handling.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Forecast: No further work is planned for this study.

3.13.11 Study 8.3.1.17.4.3 Quaternary Faulting Within 100 km of Yucca Mountain, Including the Walker Lane

The objective of this study is to identify Quaternary faults within 100 km of Yucca Mountain; and to characterize faults capable of future earthquakes with magnitude such that associated ground shaking could impact design or affect performance of the waste facility.

Activity 8.3.1.17.4.3.1 - Conduct and evaluate deep geophysical surveys in an east-west transect crossing the Furnace Creek fault zone, Yucca Mountain, and the Walker Lane. The primary objectives of this activity are to provide geophysical data and analysis that will (a) help to identify, locate, and characterize potentially significant seismic source zones; (b) characterize the crustal velocity structure and define lateral inhomogeneities in that structure; and (c) assist in determining whether buried magma bodies are present in the Yucca Mountain area.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.3.2 - Evaluate Quaternary faults within 100 km of Yucca Mountain. The primary objectives of this activity are to establish the abundance, distribution, and geographic orientation of known and suspected Quaternary faults within 100 km of the potential repository site, and to characterize those faults within this area whose apparent length or recurrence rate indicate a potential for future earthquakes of sufficient magnitude to affect design or performance of the waste facility.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.3.3 - Evaluate the Cedar Mountain earthquake of 1932 and its bearing on wrench tectonics of the Walker Lane within 100 km of the site. The objective of this activity is to evaluate the relevance of the 1932 Cedar Mountain earthquake to potential sources of ground shaking and rupture in that part of Walker Lane within 100 km of Yucca Mountain.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.3.4 - Evaluate the Bare Mountain fault zone. The objectives of this activity are (a) to evaluate the potential for ground shaking associated with future movement along the Bare Mountain fault zone; (b) to estimate the age of the most recent faulting on the Bare Mountain frontal fault; (c) to estimate the recurrence intervals of faulting; (d) to determine the nature and age of faulting within the fault complex east of the frontal zone, (e) to determine the nature of tectonic control of the location and orientation of the main wash in Crater Flat; and (f) to determine the subsurface configuration of fault zones.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.3.5 - Evaluate structural domains and characterize the Yucca Mountain region with respect to regional patterns of faults and fractures. The objectives of this activity are to map faults and lineaments within 100 km of the site and identify those with geomorphic expression indicative of Quaternary faulting; to classify the area into subareas (domains) containing relatively homogeneous faults and lineaments; to map the areal extent of desert varnish coating; and to identify areas of suspected hydrothermal alteration.

No progress was made during the reporting period; this was an unfunded activity

Activity 8.3.1.17.4.3.6 - Analyze rotation (drag) of bedrock along or over suspected wrench faults based on rotation of paleomagnetic declinations. The objective of this activity is to determine the spatial and temporal patterns of oroflexure bending based on rotation of paleomagnetic declinations.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Forecast: No further work is planned for this study.

3.13.12 Study 8.3.1.17.4.4 - Quaternary Faulting Proximal to the Site Within Northeast-Trending Fault Zones

The objective of this study is to evaluate the potential for ground motion resulting from future movement of Quaternary strike-slip faults east and south of the site area.

Activity 8.3.1.17.4.4.1 - Evaluate the Rock Valley fault system. The objectives of this activity are (a) to determine the location, spatial orientation, length, width, Quaternary recurrence rate, and the location, amount, and nature of Quaternary movement of the Rock Valley fault system; and (b) to estimate the total displacement, including strike-slip and dip-slip components, of Quaternary datums.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.4.2 - Evaluate the Mine Mountain fault system. The objective of this activity is to determine the location, spatial orientation, length, width, Quaternary recurrence rate, and the location, amount, and nature of Quaternary movement of the Mine Mountain fault system.

This activity is complete. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.4.3 - Evaluate the Stagecoach Road fault zone. The objective of this activity was intentionally deleted.

The objective of this activity has been met under Study 8.3.1.17.4.6 (see Section 3.13.14 of this progress report).

Activity 8.3.1.17.4.4.4 - Evaluate the Cane Spring fault system. The objective of this activity is to determine the location, spatial orientation, length, width, Quaternary recurrence rate, and the location, amount, and nature of Quaternary movement of the Cane Spring fault system.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Forecast: No further field work is planned for this study.

3.13.13 Study 8.3.1.17.4.5 - Detachment Faults at or Proximal to Yucca Mountain

The objectives of this study are to supply information pertaining to distribution, displacement rate, and age of detachment faults proximal to Yucca Mountain. Key questions regarding detachment faults are whether they represent a significant earthquake source, and whether they conceal a significant earthquake source at depth. To resolve both questions, activities are focused on resolving the Quaternary behavior of postulated detachment faults.

Activity 8.3.1.17.4.5.1 - Evaluate the significance of the Miocene-Paleozoic contact in the Calico Hills area to detachment faulting within the site area. The objectives of this activity are to determine whether the contact of Miocene volcanic rocks on Paleozoic strata is tectonic or depositional; if tectonic, to determine Quaternary activity, if any, of the possible detachment fault; and, if Quaternary, to determine the direction and age of movement, attitude of fault plane, and nature of deformation of the Miocene (upper plate?) sequence.

This activity is complete. See Progress Report #13 (DOE, 1996f).

Activity 8.3.1.17.4.5.2 - Evaluate postulated detachment faults in the Beatty-Bare Mountain area. The objective of this activity is to determine if postulated detachment faults in the Beatty Bare Mountain have been active in the Quaternary.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.5.3 - Evaluate the potential relationship of breccia within and south of Crater Flat to detachment faulting. The objective of this activity is to determine whether breccias tectonically emplaced on low-angle surfaces beveled across the Paleozoic and younger strata are slide masses or near-surface parts of a detached upper plate; and, if either, how they relate to postulated Quaternary detachment faulting.

This activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.5.4 - Evaluate postulated detachment faults in the Specter Range and Camp Desert Rock areas. The objective of this activity is to determine whether the basal contact of the Horse Spring Formation is depositional or tectonic; and, if tectonic, to determine whether movement was Quaternary or older, and if Quaternary, to determine the direction and amount of offset, the amount of extension, and the style of extensional deformation of the upper plate.

This activity is complete. See Progress Report #13 (DOE, 1996f).

Activity 8.3.1.17.4.5.5 - Evaluate the age of detachment faults using radiometric ages. The objectives of this activity are to determine if the subdetachment basement and the Bare Mountain massif cooled through the blocking temperatures of zircon and apatite during the Quaternary Period, and to determine if the Northern Amargosa core complex cooled through the blocking temperatures of muscovite and biotite during the Quaternary Period.

This activity was canceled because active detachment faulting is not present at Yucca Mountain. See Progress Report #15.

Forecast: No additional work is planned for this study.

3.13.14 Study 8.3.1.17.4.6 - Quaternary Faulting Within the Site Area

The objectives of this study are to identify and characterize Quaternary faults that intersect or project toward the surface facility, repository, or controlled area; and to identify and characterize Quaternary faults at the site whose length or recurrence rate suggest a potential for future earthquakes with magnitudes such that associated ground shaking could affect design or performance of the waste facility.

Activity 8.3.1.17.4.6.1 - Evaluate Quaternary geology and potential Quaternary faults at Yucca Mountain. The objectives of this activity are to synthesize and evaluate data pertaining to location, orientation, length, width, Quaternary recurrence rate, and location, amount, and nature of Quaternary movement of faults within the site area; and to identify unrecognized faults in the site area.

See the description under the following activity 8.3.1.17.4.6.2).

Activity 8.3.1.17.4.6.2 - Evaluate age and recurrence of movement on suspected and known Quaternary faults. The objectives of this activity are (a) to determine through trenching and trench wall mapping the location, spatial orientation, length, width, Quaternary recurrence rate, interconnections at the surface, and the location, amount, and nature of Quaternary movement of the Windy Wash, Solitario Canyon, Ghost Dance, and Paintbrush Canyon faults and other suspected or possible Quaternary faults within the site area; and (b) to determine through trenching and dating the age, amount, and nature of offset and the recurrence history of the Bow Ridge fault system and to evaluate that information in context with data contributed by other studies on the age, nature, and origin of fracture coatings and fissure fillings deposited within that zone.

Nearly all the work for this study was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Forecast: Any additional work deemed necessary will be performed under activity 8.3.1.17.3.6.2.

3.13.15 Study 8.3.1.17.4.7 - Subsurface Geometry and Concealed Extensions of Quaternary Faults at Yucca Mountain

The objectives of this study are to provide data on the distribution of mass, magnetic gradients, geoelectric features, and seismic velocities and reflections that will aid in evaluating the continuity of Quaternary faults where concealed by Holocene and late Pleistocene surficial deposits, to evaluate the data and its limitations, to evaluate the possibility that Quaternary faults exposed as high-angle faults at the site continue to depth as planar, high-angle faults, or alternatively, flatten at depth and merge with one or more long-angle faults; and to provide information on continuity of rock units within the repository and controlled area to assist the investigation of site geology.

There will be no study plan developed for this study and the eight associated activities. Field geophysical surveys and analysis of the subsurface geometry and concealed extensions of Quaternary faults are to be performed under Study 8.3.1.4.2.1 discussed in Section 3.3.3 of this progress report. Geophysical surveys conducted for Study 8.3.1.4.2.1 will be examined as inputs for assessing concealed faults and subsurface geometries. The implications of subsurface geometry and concealed extensions of Quaternary faults will be addressed as part of the evaluations associated with Study 8.3.1.17.3.6, discussed in Section 3.13.8 of this progress report.

Forecast: No additional work is planned under this study.

3.13.16 Study 8.3.1.17.4.8 - Stress Field Within and Proximal to the Site Area

The objective of this study is to provide data on ambient stress at the site and its immediate vicinity that will aid in evaluating most favored orientation and nature of future movement on faults within the site area, stability of potential pathways for radionuclide travel controlled by or related to fracture aperture, the stability of mined excavations, response of rock mass to thermal loading, and applicability of tectonic models. A secondary objective is to evaluate the potential relevance of paleostress data to prediction of future stress orientations.

As originally described in the SCP, four activities were included in Study 8.3.1.17.4.8, but because (a) the objectives and parameters assigned to two of the activities are very similar, and (b) tasks assigned to the other two activities are adequately covered by Study 8.3.1.17.4.12 (Section 3.13.20 of this progress report), the scope of this study is now limited to one activity.

Activity 8.3.1.17.4.8.1 - Evaluate Present Stress Field Within and Proximal to the Site Area. The objective of this activity is to measure the vertical and lateral variation of in situ stress at and proximal to the potential repository by conducting hydraulic fracturing stress measurements and observations of stress-induced borehole breakouts in boreholes that are scheduled to be drilled adjacent to the site. The magnitudes and orientations of the horizontal and vertical in situ stresses are the principal parameters to be determined.

Previous hydrofracture stress measurements are described in Stock et al. (1985). Interferences concerning the orientation of the stress field based on earthquake focal mechanisms are summarized in Whitney (1996, Chapter 7) (see Progress Report #15, DOE, 1997e). New hydrofracture stress measurements performed in the ESF as part of the In-Situ Design Verification program are described in Section 3.11.8.

Forecast: Remaining work for this study is contingent upon the drilling of new boreholes.

3.13.17 Study 8.3.1.17.4.9 - Tectonic Geomorphology of the Yucca Mountain Region

The objective of this study is to document Quaternary uplift and subsidence within the Yucca Mountain region and to evaluate regional variation in the nature and intensity of Quaternary faulting.

There will be no study plan developed for this SCP section and no work is being planned. The work scope was transferred to and performed under Studies 8.3.1.5.1.4 and 8.3.1.17.4.12 (Sections 3.3.4 and 3.13.20, respectively, of this progress report). Data to describe the characteristics and parameters of tectonic features were collected under Studies 8.3.1.17.4.3 and 8.3.1.17.4.6 (Sections 3.13.11 and 3.13.14, respectively, of this progress report).

Forecast: No additional work is planned under this study.

3.13.18 Study 8.3.1.17.4.10 - Geodetic Leveling

The objective of this study is to evaluate possible historical and contemporary vertical displacements across potentially significant Quaternary faults within 100 km of Yucca Mountain. Secondary objectives are (a) to characterize the historical rate of uplift and subsidence in the Yucca Mountain region and (b) to evaluate the possible existence of tectonic boundaries, coinciding perhaps with the Walker Lane or with the Furnace Creek fault zone, that may separate domains with differing rates of uplift and subsidence.

Three activities are assigned to Study 8.3.1.17.4.10: Activity 8.3.1.17.4.10.1 (Relevel base-station network, Yucca Mountain and vicinity); Activity 8.3.1.17.4.10.2 (Survey selected base stations, Yucca Mountain and vicinity, using global positioning satellite); and Activity 8.3.1.17.4.10.3 (Analyze existing releveling data, Yucca Mountain and vicinity). Because the work involved in these activities is closely interrelated, the activities have been combined.

No progress was made during the reporting period; this was an unfunded study. See Appendix A, Section A.1.14.4 of this progress report for additional information.

Forecast: No work is planned in FY 1997.

3.13.19 Study 8.3.1.17.4.11 - Characterization of Regional Lateral Crustal Movement

The objective of this study is to evaluate rates and orientation of historical and current crustal strain based on analysis of existing data on seismicity, historical fault, offset, and creep in the Basin Range and at Yucca Mountain.

No unique data are required by this study. All activities were transferred to Study 8.3.1.17.4.10 (see Section 3.13.18 of this progress report).

Forecast: No further work is planned under this study.

3.13.20 Study 8.3.1.17.4.12 - Tectonic Models and Synthesis

The objectives of this study are (a) to synthesize data relevant to tectonics; (b) to develop a model or range of models that establishes the causal relation between application of tectonic forces and formation of structures observed at Yucca Mountain and vicinity, to link observed rates of formation of structures with regional rates of crustal strain; (c) to forecast changes in tectonic setting and the manner in which changes will affect both regional crustal strain rate and tectonic stability in the Yucca Mountain region; (d) to estimate effects of changes on rates and nature of crustal strain at Yucca Mountain and vicinity, and (e) to estimate future rates of tectonic processes at Yucca Mountain.

Activity 8.3.1.17.4.12.1 - Evaluate tectonic processes and tectonic stability at the site. The principal objectives of this activity are to synthesize geologic and geophysical data pertaining to faults (normal, detachment, strike-slip) and other structural and volcanic features in the Yucca Mountain area; and to evaluate these data in terms of the tectonic stability of the potential repository site. For additional information on the objectives, see Study Plan 8.3.1.17.4.12.

See description under Activity 8.3.1.17.4.12.2.

Activity 8.3.1.17.4.12.2 - Evaluate tectonic models. The objectives of this activity are (a) to formulate a range of tectonic models that relate the nature and estimated rates (including bounding values) of Quaternary processes (volcanism, faulting, uplift, and subsidence, lateral strain, and possibly folding) of potential significance to design and performance of the repository at Yucca Mountain; (b) to evaluate temporal changes in tectonic activity and resulting changes in fractures and other structural features of potential hydrologic significance at and in the vicinity of Yucca Mountain (relate tectonic cycle, if it exists, to tectonic model(s)); (c) to ensure that assumptions, inferences, and conclusions concerning tectonic processes that are important to design and performance of the repository are consistent with tectonic models applicable to the site; and (d) to ensure that uncertainty in the data, assumptions, and inferences concerning rates and nature of those tectonic processes that are important to design or performance of the repository is adequately reflected in conclusions about those processes.

Most of the work for this activity was completed in FY 1996. See Progress Report #15 (DOE, 1997e).

Activity 8.3.1.17.4.12.3 - Evaluate tectonic disruption sequences. The objective of this activity is to evaluate disruption sequences involving faulting, folding, uplift and subsidence, and volcanism that are of potential significance to design or performance of the repository.

Work to accomplish the objective of this activity is being performed under Study 8.3.1.8.2.1 Tectonic Effects: Evaluation of Changes in the Natural and Engineered Barrier Systems Resulting from Tectonic Processes and Events. Progress is discussed in Section 3.6.3.

Forecast: Any additional work deemed necessary for this study will be conducted under Study 8.3.1.8.2.1 (Section 3.6.3 of this progress report).

3.14 STUDY 8.3.1.20.1.1 - ALTERED ZONE CHARACTERIZATION

The Altered Zone Characterization Program was not included in the SCP. However, the changes to the program are summarized in Appendix A, Section A.1.15.

The objective of this study is to characterize the effects on the region around the potential repository that is altered by hydrothermal processes that develop in response to heating of the repository block due to radioactive decay of emplaced nuclear waste.

<u>Activity 8.3.1.20.1.1.1 - Field and laboratory studies of the effects of mineralogical and mechanical changes on transport processes</u>. The objective of this activity is to evaluate the impact of chemical, mineralogical, and mechanical change on hydrological properties, particularly on porosity and permeability, and the kinetics of these processes, as functions of several environmental variables, including temperature, fluid composition, fluid flow rate, stress, and water volume to surface area ratio. Also considered in these studies will be the relationship between pore geometry and permeability as recrystallization occurs within crushed, fractured, and intact materials.

Previous plug-flow reactor experiments (a) established the overall protocol for conduct of these experiments, and (b) demonstrated the ability to simulate reactive transport in simple chemical systems (see Progress Report #15, Section 3.14, DOE, 1997e). During this reporting period, plug-flow reactor experiments have been designed to consider reactions in complex geological media similar to that near the repository horizon. One of these experiments has been completed and successfully modeled, using the reactive-transport code GIMRT (Global Implicit Multicomponent Reactive Transport). (See Table 6-1 in Chapter 6 for a description of the code). This work was done to establish reaction kinetics constraints for future simulations that will use more reactive vitric material.

In the first experiment (Johnson et al. 1997), Tsw2 rock was pulverized to 125 to 75 micron size fraction, placed in a titanium cylinder 3.1-cm long with a 0.66-cm effective diameter, and infiltrated by distilled water at a flow rate of 25 mL/day for 36 days at 240°C and 84.1 bar. As reacted fluid exited the plug-flow reactor du[illegible]gressive dissolution of the crushed tuff, concentrations of the major cations (calcium, so[illegible], potassium, silicon, aluminum, and magnesium) were systematically monitored; these attained approximately steady-state values within 3 to 4 days. [Note: Although the plug-flow reactor experiment was performed at 240°C and 84.1 bar, the thermodynamic data base used in the GIMRT simulation analog is constructed for $P_{sat}(T)$, which at 240°C is 33.4 bar. This pressure discrepancy, however, is insignificant.]

The solid reaction products were examined using standard x-ray diffraction techniques, and scanning electron microscopy. The results demonstrate clear textural evidence consistent with dissolution of primary cristobalite and feldspars at the inlet of the reactor, which is consistent with the results from the simulations. At the outlet, dissolution of feldspars was apparent as well, although the extent of dissolution was significantly less than that observed in the inlet. It was also evident that cristobalite precipitation had begun, which was consistent with the observation in the water analyses that cristobalite saturation had been reached.

This experiment was used to determine the appropriate dissolution and precipitation kinetics parameters for the phases involved in reactive transport. The crushed tuff was used to evaluate the suitability of existing models of reaction kinetics, where the data was generated over a short time period at laboratory scale. Other experiments using fractured and intact materials are a means to determine the suitability of scaling process models from crushed materials to mountain scale. Primary consideration was given to the dissolution kinetics of the primary feldspar component (K-feldspar, albite, anorthite), and the precipitation kinetics for the possible secondary phases (quartz, cristobalite, muscovite, paragonite, kaolinite, pyrophyllite, gibbsite, diaspore, and boehmite). Varying the reaction rate parameters for these phases, within the uncertainty limits of the measured values for the rate constants and for the activation energies, allowed the results of the experiments to be accurately reproduced. Rate constants most suitable for modeling evolution of these rocks were obtained.

Activity 8.3.1.20.1.1.2 - Evaluating existing and developing future capabilities to simulate coupled hydrothermal and reactive transport processes. The objectives of this activity are to compare codes and to evaluate their suitability for application to altered zone efforts in two steps. The first step will be to review the capabilities of the codes as they currently exist. Comparisons will be made of how the codes simulate well-documented processes. Once several codes are selected, they will be used to simulate the results of the ongoing experimental studies and to predict the outcomes of those studies. They will also be used to simulate field properties of sites selected for field study. These forecasts will be used to refine modeling strategies, as discrepancies between measurement, observation, and field data become evident.

Extensive simulation efforts indicate that the OS3D/GIMRT and NUFT code packages provide the most suitable capabilities to model the processes of concern, among the code packages considered. Appendix I describes these codes, which are now in use to evaluate a variety of reactive transport concerns.

During the reporting period, the OS3D/GIMRT code package was further tested by considering specific numerical effects on the calculational results. Preliminary tests were made of different computational modes, using similar initial input. Interest here was on determining whether significant discrepancies appear in the results, if input options are exercised for concentration limits, and if constraints are placed on extent of supersaturation to control precipitation kinetics. Results to date indicate little effect on the results of the calculations.

Activity 8.3.1.20.1.1.3 - Performing bounding calculations of the effect of coupled processes in the altered zone on near-field properties. The objective of this activity is to determine parameter values, limits, or ranges needed to define the waste package environment.

Simulations were conducted to define the conditions under which dissolution and/or precipitation would impact matrix and fracture porosities and permeabilities. Reactive transport simulations were conducted for flow regimes that may be expected near the boiling front under conditions in which saturation has been achieved. Preliminary results demonstrate that the primary concern for hydrological properties is the formation of cristobalite and/or calcite plugs in fractures in which fluxes exceed approximately 100 mm per year. These preliminary results show that under these conditions, precipitation of solids at the boiling front have the potential to seal fractures in less than 100 years. These results will be refined as further simulations are conducted.

Activity 8.3.1.20.1.1.4 - Performing bounding calculations of the effect of coupled hydrological and reactive transport processes on thermal evolution. The objective of this activity is to determine the response of the altered zone over time using, as initial conditions for simulations, optional designs, and operating configuration of the potential repository.

The effort this year is focused on establishing the location and magnitude of changes in hydrologic properties that could influence heat transfer. The results described for Activity 8.3.1.20.1.1.1 (using the reactive transport simulator GIMRT) were used (a) to establish reaction rates for mineral phases important for changes in hydrological properties that control heat and mass transfer, (b) to predict potentially verifiable porosity and mineralogic evolution during reaction between idealized TSw2 rock and infiltrating fluid compositions, and (c) to model unverifiable, long-term porosity evolution in the post-emplacement altered zone. The results of these models can be provided to thermal-hydrological codes, such as NUFT, to allow iterative updating of flow fields, in order to bound the effects of dissolution and precipitation on repository thermal evolution.

The results of the simulations mapping mineralogical changes are being incorporated into simulations of the thermal evolution of the near-field environment and the altered zone. Preliminary results suggest that fluid movement may be restricted by plugging of fractures at the boiling front, in regions where movement of the boiling front is slow. These results also suggest fluid may drain through pillars between emplacement drifts due to condensation as water enters regions of lower temperatures. Further modeling of fluid transport to regions below the emplacement drifts is in progress, to evaluate the extent to which flow barriers may form because of porosity changes associated with the development of alkaline plumes below emplacement drifts, caused by water interacting with cementitious invert materials.

These initial results are preliminary, and the predictions require considerable refinement.

Forecast: Descriptions of the conditions under which barriers or preferential flow pathways may form will be determined and documented in a format useful for total system performance assessment. Activities will concentrate on conducting detailed reactive transport simulations using repository-relevant properties and geometries. The simulations will be tested using a small suite of experiments to verify the applicability of the simulation results. Activities will focus on establishing realistic reaction rates within the bounds of experimentally measured rates.

CHAPTER 4 - REPOSITORY DESIGN

INTRODUCTION

The Yucca Mountain Site Characterization Project (Project) is presently engaged in a comprehensive site investigation program to confirm the suitability and adequacy of Yucca Mountain as a potential geologic site for high-level radioactive waste disposal. The repository design activities are being conducted consistent with the eventual design, licensing, and construction of a Mined Geologic Disposal System (MGDS) (commonly called a geologic repository).

The design effort continued in support of major upcoming Civilian Radioactive Waste Management Site Characterization Program (Program) milestones (viability assessment, environmental impact statement, site recommendation, and license application). The design effort, which is building on the work performed for the advanced conceptual design, was summarized in the introduction to Chapter 4 of Progress Report #15 (DOE, 1997e).

Design Basis Event Analysis

The Project continues to better define the level of design detail required to support the viability assessment and the level needed for license application design. A major effort supporting this process involved refining and analyzing the preliminary set of design basis events developed last period. The analysis of design basis events will also be used to refine the list of repository and waste package systems, structures, and components subject to quality assurance (QA) requirements. The 22 external events (caused by factors not directly related to repository design or operation) selected for further analysis last reporting period were grouped into 11 potential analysis groups. The groups were then prioritized for analysis on the basis of potential impact on repository design, availability of information to support the analysis, and whether the analysis is needed to support the viability assessment. Likewise, internal events (those caused by factors directly related to repository design or operation) were grouped into 16 potential analysis groups. Two pilot analyses were begun that will serve as templates for other design basis event consequence analyses that will be performed in future reporting periods.

Design Approach

The Project continues to develop a design to support the 1998 viability assessment. The design will continue to evolve to support the environmental impact statement, the site recommendation, and the license application that will be submitted following the viability assessment. Efforts will focus directly on (a) supporting the major Program milestones, (b) developing and identifying a current design to support postclosure performance evaluations, (c) addressing likely regulatory issues associated with first-of-a-kind aspects of the repository and waste package designs, and (d) refining estimates of the costs of the designs. The MGDS design will continue to evolve to a level of detail that will be adequate to support submittal of a license application, but not to a level of detail adequate for construction. The design to support construction will be developed later. The design approach was described more fully in the introduction to Chapter 4 in Progress Report #15.

The Project's understanding of the site continues to evolve. Therefore, repository and waste package designs are continually being reassessed in light of this knowledge and of the waste containment and isolation strategy being developed to achieve acceptable repository performance. Changes in estimated percolation flux are of particular importance in these reassessments. Data generated during the latter part of this reporting period suggests that percolation flux at the repository emplacement horizon may be significantly higher than previously estimated. Even though there has always been a design incentive to keep the waste packages dry, the updated flux information has added emphasis to the importance of enhancing site performance by use of robust, more durable engineered barriers.

Repository design activities have been adjusted to focus more attention on engineered barrier design options. Efforts are in progress to identify and evaluate a variety of options in terms of their relative merit for performance improvements and for their costs. These efforts are being conducted in conjunction with performance assessment and scientific staff using total system performance assessment models, two-dimensional and three-dimensional near-field models, and design basis models.

Support for Viability Assessment

As stated in Progress Report #15, the design supporting the viability assessment will build on existing design work documented in the MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b) and will emphasize the key technical questions that affect waste containment and isolation, performance, and cost. Design elements critical to determining the feasibility and performance of the repository and the engineered barrier system will be emphasized, as will systems and components that may be significant cost drivers for the repository. The effort will evaluate the technological feasibility of the designs of selected systems, structures, and components, but will generally not develop the design detail needed to submit the license application. The key cross-cutting design issues for which progress toward resolution must be achieved to have an adequate design to support the viability assessment are listed below.

- Thermal loading range
- Engineered barrier system performance enhancements
- Criticality control
- Emplacement drift ground support concept
- Performance confirmation concept
- Retrievability concept

- Confirmation of high volume and long period waste handling capability and design basis events consequences
- Disposal of site-generated waste
- Strategy for mapping repository subsurface
- Postclosure performance standards
- Viability of underground remote control concepts
- Repository seals requirements and concepts
- Regional Service Agent/Interin Storage Facility interface
- Additional waste forms
- Waste package sizes and weights
- Waste package materials
- Design basis model
- Subsurface development
- Surface development
- Site development.

These issues do not need to be completely resolved to support the viability assessment. However, because the plan to design structures, systems, and components to support the submittal of a license application is based on the concept of a "one-pass" design (that is a design that is intended to support license application and repository construction), enough progress must be made to minimize cost and schedule issues associated with redesign.

Scientific investigations and performance assessment activities will provide data and requirements for the design effort. Applicable data will come from scientific and engineering tests and analyses in the Exploratory Studies Facility (ESF), as well as from surface-based and laboratory testing and analysis. In turn, the updated total system performance assessment in support of the viability assessment will use the updated design concepts and the analyses of available site and engineering data. The primary objective of the total system performance assessment is to evaluate the probable behavior of the potential repository in the Yucca Mountain geologic setting. An additional objective is to further refine evaluations of repository performance under a range of normal conditions and under conditions imposed by potentially disruptive events, such as earthquakes and volcanism. The performance assessment will also evaluate the possible range of performance caused by uncertainty in estimating key factors such as ground-water flow, thermal effects, and corrosion.

Establishing the Regulatory Basis

In conjunction with the effort in support of the viability assessment, the Project began an effort that will provide regulatory guidance to support the eventual license application. Appropriate acceptance criteria and applicable U.S. Nuclear Regulatory Commission (NRC) regulatory and licensing precedent will be identified for Bin 3 (important to safety or waste isolation and little or no NRC regulatory precedent) and Bin 2 (important to safety or waste isolation with NRC regulatory precedent) systems, structures, and components. The binning (categorization) process was explained in the introduction to Chapter 4 of Progress Report #15. Basically, it entails classifying systems, structures, and components for design priority using the importance of the items to safety or waste isolation and whether or not regulatory precedent exists for them. The more important items with less regulatory precedent will be designed in greater detail and probably earlier in the design effort to support the license application than will less important items or those with more regulatory precedent.

The regulatory guidance for the design organization will be documented in an Engineering Compliance Plan. This plan will also provide acceptance criteria that will be used by the authors of the engineering chapters of the eventual license application, the systems description documents, the engineering design guides, and the design bases event analyses. The plan will identify the information necessary to provide reasonable assurance to the NRC that the repository design supports construction of a repository that would not pose an unacceptable risk to the health and safety of the public or repository workers. The effort will include identification of NRC regulatory guidance, licensing precedent, and industry standards that may be applicable to the repository design. Additional regulations, NRC Regulatory Guides, and NRC NUREG documents may be invoked when additional guidance is needed beyond that provided in 10 CFR Part 60.

The Engineering Compliance Plan will be developed in a phased manner to support the phased design process described in Progress Report #15. (Phase I design will support the viability assessment.) Thus, the regulatory guidance applicable to the design of Bin 3 and Bin 2 systems, structures, and components to be designed as part of the Phase I design will therefore be developed first. Then the guidance to support Bin 3 and Bin 2 systems, structures, and components for phase II design (to support the license application) will be developed. In addition to supporting development of the license application, the Engineering Compliance Plan will be used to focus the design work in progress on meeting the regulatory and other requirements and standards considered applicable to the repository. If regulatory requirements change, the engineering compliance program will be revised accordingly.

Design Activities This Reporting Period

Design activities focused on continued development and refinement of the repository design concepts that will support the viability assessment and the license application. Highlights included the following:

- The reference three-dimensional thermal/mechanical stratigraphy model of Yucca Mountain was updated to reflect ever-increasing availability of site information. This information is used to support design (see Section 4.1.3).

- The repository design and emplacement concepts continued to be refined to minimize the amount of excavation necessary to emplace waste. Analysis of areal mass loading indicated that raising the loading to about 85 MTU/acre would slightly reduce (two percent) the repository area required from that required in the advanced conceptual design while still meeting thermal goals. Likewise, minimizing drift space surrounding defense high-level waste could reduce emplacement area required by about 10 percent. Finally, using wider drift spacing coupled with closer waste package spacing in the drifts could also be used to reduce the amount of excavation needed. Work in these areas continues, and the concepts have not yet been approved for implementation in the design (see Sections 4.1.6 and 4.1.17).

- Preliminary quantitative guidelines for use of tracers, fluids, and other materials in emplacement drifts were developed to help ensure such materials pose no unacceptable risk for repository performance. In addition, a list of information needed on planned use of such materials was developed to support performance assessment activities (see Section 4.1.12).

- Concepts for the repository exhaust main were refined (see Section 4.1.14).

- Concept development and initial drawing layouts were made for observation drifts to support the performance confirmation program after waste emplacement (see Section 4.1.14).

- Examination of the benefits of emplacement drift backfill to repository performance continued. A study of the potential benefits of backfill in reducing relative humidity at the waste package found no appreciable performance improvement from use of backfill. Future analysis will need to consider potential capillary effects of backfill and mechanical protection provided by backfill against waste package damage from rockfall to enable a decision to be made on use of backfill (see Section 4.1.18).

- Development continued on design analyses that may affect repository operations concepts. For example, a radiological safety design analysis is in progress that will estimate personnel exposure to radiation in waste handling operations in the repository surface facilities. Also, a retrieval design analysis is in progress that will provide concepts for equipment needed, failure modes of that equipment, and activities associated with retrieval of waste. In other progress, a design analysis demonstrated that it is feasible to use a remote-controlled gantry to travel through an emplacement drift and obtain data for performance confirmation without first having to cool the emplacement drift. Such a system would result in better data being acquired and also less thermal disturbance on the emplacement drift and waste packages (see Section 4.4.2).

- Development was begun of design guides for radiation, drifts, source terms, and remote operations. These design guides will provide the design methodology and detailed design criteria for each of these subjects.

- Rail alignments analyses and study of the boundaries of an additional potential transportation corridor beyond the four already under consideration were completed to support work on waste transport within the state of Nevada but outside the site boundaries.

- Program cost estimates were completed for 99 000 and 70 000 MTU emplacement scenarios. Cost-estimating spreadsheets were upgraded in preparation for the cost estimates required for viability assessment. Cost estimates were made to support the various repository trade studies and design analyses. The 99 000 MTU represents the total waste to be disposed of by the program, while the 70 000 MTU represents the waste to be disposed of at the proposed Yucca Mountain repository.

- Analyses were nearly completed on subsurface layout, subsurface coordinate geometry, and subsurface construction and development methodology. These will define the subsurface repository configuration, construction sequence, and construction equipment for viability assessment.

- The Waste Handling Systems Configuration Analysis (CRWMS M&O, 1997d) was completed. The analysis established the number of operating lines and capacity of in-process staging areas for waste handling operations and recommended a preferred waste handling technology. The number of lines would be five (three wet and two dry), and the capacity of the staging area would be 78 fuel assemblies per line. The wet lines will handle transfer of spent nuclear fuel assemblies, while the dry lines will handle canistered waste forms (such as disposable containers and defense high-level waste canisters).

- The Operations Staffing Letter Report was completed. The report described the results of a limited analysis that developed preliminary operating strategies and staffing estimates for repository operations. A rough estimate for required repository operating staff, based on the advanced conceptual waste handling design, was 358 for the day shift, 151 for the second shift, and 147 for the third shift. Total site staffing, which includes personnel for subsurface development but excludes subcontract labor, was estimated to be 609 for the day shift, 341 for the second shift, and 211 for the third shift.

- Development began of the following products to support surface design: radiological safety design analysis, recovery operations analysis, space allocation analyses for waste handling systems, mechanical flow diagrams for the material handling systems, and process flow diagrams for waste treatment and pool water treatment systems. These documents, when completed, will all support advancement in development of various aspects of the surface design.

Repository Consulting Board

In June 1996, at the Project's suggestion, the ESF Tunneling Board of Consultants changed its focus from ESF activities to the repository program. As a result, the board is now called the Repository Consulting Board. The Board's charter is to provide recommendations to the Project on the following aspects of repository design:

- Ground support
- Tunnel stability
- Constructability and operability
- Waste emplacement and retrievability.

The Board met two times this reporting period: December 4-5, 1996 and February 20-21, 1997.

During these meetings, the Board was briefed on a range of repository subsurface design topics, including construction methodology and sequence, ventilation, performance confirmation plans, and ground support. The Board has provided valuable feedback and guidance to the ongoing subsurface design, and it is currently scheduled to meet again April 24-25, 1997.

4.1 CONFIGURATION OF UNDERGROUND FACILITIES (POSTCLOSURE) (SCP SECTION 8.3.2.2)

4.1.1 Design Activity 1.11.1.1 - Compile a Comprehensive List of All the Information Required From Site Characterization

The objective of this design activity is to summarize, in one document, all the information required from site characterization for repository surface and subsurface design. This information will be acquired before or during design. Program flexibility will exist such that design can proceed, even in the absence of certain information identified as required, by making suitable assumptions that will require verification.

Design data needed from site characterization continued to be identified for various uses, including the following:

- Design data needs, which have been identified in the Repository Design Data Needs report (CRWMS M&O, 1995b), were used as a basis for developing specific testing and data gathering activities for the ESF drift-scale heater test. These drift-scale testing activities, which include the measurement of thermally induced temperature, rock deformation, and ground support deformation, are described in two documents (CRWMS M&O, 1997e; DOE, 1997f). Additional geomechanical data that are especially important to design include the rock mass thermal expansion coefficient and the rock mass modulus. These data and their spatial distribution were identified for data gathering activities in the ESF in fiscal year (FY) 1997.

- Seismic design data needs, which include vibratory ground motion and fault displacement parameters, were identified in the Repository Design Data Needs report (CRWMS M&O, 1995b). These data needs have been identified for detailed data gathering activities in support of the current Probabilistic Seismic Hazards assessment that will produce a seismic design report in early FY 1998.

- Data needs for concrete material properties, specifically strength, modulus, and creep at expected repository temperatures, have been identified for laboratory testing planned for FY 1997. Additional laboratory tests have been planned to examine temperature-induced chemical reactions in concrete that may influence the pH of the long-term concrete and ground-water system.

- Geologic data to provide improved stratigraphic control[1], especially of the southwest portion of the repository block, and geotechnical data related to the prediction of rock stability have been identified for measurement by possible borehole activities in FY 1997. These data would be obtained partly by means of open hole drilling and downhole geophysical logging and partly by coring.

__Forecast__: The Repository Design Data Needs report (CRWMS M&O, 1995b) will continue to support project planning and to provide the basis for the development of more detailed data gathering activities. Continued design activities may result in refined data needs, as well as elimination of other data needs. Changes to data needs will continue to be provided to the Test Coordination Office for disposition. Revisions to the Data Needs Report are not planned.

4.1.2 __Design Activity 1.11.1.2 - Determine Adequacy of Existing Site Data__

The objective of this design activity for the U.S. Department of Energy (DOE) is to determine whether the available site data are sufficient for licensing. (The ultimate determination of sufficiency will be made by the NRC.) If not, it will be determined whether additional data must be gathered or whether the design can be completed using assumed

[1]Stratigraphic control is the process by which stratigraphy is predicted at locations other than at known data points (locations), referred to as "control points." Stratigraphic control is improved by acquiring more control points.

parameters until additional data has been collected. This is an ongoing determination up to submittal of the license application.

This activity is closely related to Design Activity 1.11.1.1, discussed in Section 4.1.1 of this progress report. Revision 00 of the Repository Design Data Needs report (CRWMS M&O, 1995b), described in Progress Reports #13 (DOE, 1996f) and #14 (DOE, 1996g), contains the analyses of sufficiency of available site data. The following activities contributed to improving confidence that, at time of license application submittal, site data will be adequate.

- Activities were planned for the ESF drift-scale heater test to improve the completeness of data gathered defining drift and ground-support behavior at high temperatures. The adequacy of geomechanical data, especially rock mass thermal expansion and rock mass modulus, will be improved by plans to obtain further measurements at several locations in the ESF.

- Evaluation and review of geologic mapping data and of results from the monitoring of ground support loads from the ESF drifts continue in order to provide further information on rock classification parameters used for repository design. These data will be used in ESF design verification studies and to provide baseline data for performance confirmation.

- Results of surface geologic mapping have been used to expand and upgrade inputs to the three-dimensional geologic computer model (LYNX geology model) that defines the volume of rock mass available for repository layout development.

Forecast: The adequacy of data will be continually assessed to support ongoing design activities.

4.1.3 Design Activity 1.11.1.3 - Document Reference Three-Dimensional Thermal/ Mechanical Stratigraphy of Yucca Mountain

The objective of this design activity is to produce reports that describe the three-dimensional thermal and mechanical stratigraphy of Yucca Mountain. The description will rely on information gathered from site characterization activities and will be entered into the Reference Information Base (DOE, 1995e). The description will then be used as the reference basis for design and performance assessment. This activity is related to Design Activity 1.11.3.2, covered in Section 4.1.7 of this progress report.

During the last reporting period, initial development of a three-dimensional computer model of the repository site area was completed for the support of repository design work. This reporting period, the model was refined and updated. This model was built using the Lynx Geoscience Modeling System (LYNX), Version 3.06 (CRWMS M&O, 1995c), which is qualified for quality affecting work. The model, designated YMP.MO3, is an update from the previous model YMP.MO2 that was developed in 1995 (CRWMS M&O, 1995d). The new model incorporates outcrop information along Solitario Canyon (Buesch, et al., 1995), new drilling and revised stratigraphic picks from core logs (CRWMS M&O, 1997a), and revised surface faults (Day et al., in press). The new model covers a larger area than was presented in YMP.MO2, extending farther northward to Pagany Wash fault to enable evaluation of northward extension of the repository. The stratigraphic surfaces modeled include the following:

- Top TSw1 Thermal/Mechanical unit
- Top TSw2 Thermal/Mechanical unit
- Top Tptpmn Lithostratigraphic unit
- Top Tptpll Lithostratigraphic unit
- Top Tptpln Lithostratigraphic unit
- Top TSw3 Thermal/Mechanical unit
- Top CHn Thermal/Mechanical unit.

These units are shown in Table 4-1.

Also, the following additional features were modeled:

- Main faults
- Topography surface
- Topography minus 200-meters surface
- Topography minus 300-meters surface
- Ground-water surface
- Upper repository plane.

The LYNX three-dimensional model YMP.MO3 was used to assist in defining available repository siting area (see 4.1.7 Design Activity 1.11.3.2). The model is documented in the Determination of Available Three-Dimensional Volume for Repository Siting document (CRWMS M&O, 1997f).

The YMP.MO3 model differs from another model, the YMP reference geologic model ISM2.0 developed by the U.S. Geological Survey, in stratigraphic picks and method of extrapolation of data. Efforts were begun to reconcile the differences and arrive at a single Project geologic model.

Forecast: Application of the YMP.MO3 model to repository design will continue. Also, the adjacent expansion areas for the repository will be examined using the ISM2.0 and YMP.MO3 models.

Table 4-1. 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)

	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
	pre-Topopah Spring bedded tuff	Calico Hills Nonwelded Unit CHn	CHn1v
	Calico Hills Formation		CHn1z
Crater Flat Group	Prow Pass Tuff	Crater Flat Unit CFu	CHn2z
			CHn3z
			PPw
	Bullfrog Tuff		CFun
			BFw
			CFMn
	Tram Tuff		TRw

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4.1.4 Design Activity 1.11.1.4 - Preparation of Reference Properties for the Reference Information Base

The objective of this activity is to develop and incorporate data into the Reference Information Base (DOE, 1995e) from sources that document pertinent repository site and design properties and describe how these properties were derived.

The Reference Information Base is a controlled data base that provides summary data and information about the Project site and engineering properties. As an evolving data base, it represents the best current state of knowledge for a wide range of technical data parameters and is the primary source of approved technical reference information. The Reference Information Base provides data for engineering design and development efforts requiring specific site properties obtained from field and laboratory measurements, as well as material properties associated with waste package and engineered barrier system evaluations. For each item in the Reference Information Base, information is presented describing the data acquisition or development methodology; the statistical bases for the displayed data, including references to source data; and the qualification status of the data.

Data needs for items that will be incorporated in the Reference Information Base are identified in activities described in Section 4.1.1 of this progress report [which references the Repository Design Data Needs report (CRWMS M&O. 1995b)], in the Controlled Design Assumptions Document (CRWMS M&O, 1996c), and in the Site Characterization Plan (SCP) (DOE, 1988).

This reporting period, efforts remained focused on the review of three new data items: (1) potentiometric surfaces, (2) hydrogeologic stratigraphy, and (3) thermomechanical stratigraphy. The review was continuing at the end of the reporting period. When the review is completed, the data will be incorporated into the Reference Information Base. In addition, efforts associated with the acceptance of a data item dealing with the heat capacity properties of TSw1 and TSw2 were in progress.

Plans are underway to provide access to the Reference Information Base through the Internet for NRC and oversight group perusal, but the schedule has not yet been finalized.

Forecast: Pertinent baseline data will continue to be examined for possible incorporation into the Reference Information Base. Data items for potentiometric surfaces, hydrogeologic stratigraphy, thermomechanical stratigraphy, and rock heat capacitance will be reviewed and incorporated into the Reference Information Base. Efforts will also be focused on review and incorporation into the Reference Information Base of specific data items necessary to support the development of the viability assessment. The electronic version of the Reference Information Base will be maintained and updated as new information is received.

4.1.5 Design Activity 1.11.2.1 - Compile Waste Package Information Needed for Repository Design

The objective of this activity is to determine what waste package information is needed for the design of the underground facility, to obtain such data, and to document it in either the MGDS Requirements Document (DOE, 1996b) or in the Controlled Design Assumptions Document (CRWMS M&O, 1996c).

Table 4-2 shows the revised masses and dimensions for current waste package designs provided for repository design this reporting period. Because tolerances were not considered in determining the waste package masses and dimensions, these values are subject to change. Because of the design and construction of the container most of the clearances that will be added will tend to increase the waste package diameter rather than the length

Table 4-2. Waste Package Dimensions and Weights

	Waste Package for 21 Uncanistered Pressurized Water Reactor Assemblies	Waste Package for 12 Uncanistered Pressurized Water Reactor Assemblies	Waste Package for 44 Uncanistered Boiling Water Reactor Fuel Assemblies	Waste Package for 4 Defense High-Level Waste Canisters	Waste Package for 4 Defense High-Level Waste Canisters and 1 DOE Spent Fuel Canister
Length, mm	5335	5335	5335	3790	3790
Diameter, mm	1650	1298	1604	1785	1970
Depth of skirt, mm	225	225	225	225	225
Thickness of skirt, mm	50	50	50	50	50
Ready for emplacement weight, kg	50 423	32 236	46 424	30 511	35 692

In addition to the waste package size information, information describing a concept for waste package support design has also been provided for repository design (CRWMS M&O, 1997g). This information included a general description, structural and thermal analysis results, and engineering sketches of the design.

A modular design was developed for the waste package support assembly. A modular design is desirable because it allows flexibility in waste package placement in the drifts and individual components can be replaced if the support structure is damaged in a waste package handling accident. Because it is less critical for the support structure to be damaged than for the waste package to be breached, the support structure is designed to yield if a waste package handling accident occurs

The waste package support assembly concept consists of a waste package pier and a waste package support. (See Figure 5-5 in this progress report for an illustration of this concept.) The pier and support hold the waste package off of the invert to allow air circulation to reduce the temperature inside the waste package, and they allow for a sorbing medium below the waste packages if needed. The sorbing medium may be used to trap radionuclides after the waste package has degraded.

Two support assembly designs were included in the information provided for repository design. The first accommodates a gantry with 300-mm-diameter wheels and the second a gantry with 600-mm-diameter wheels. The main difference between the two designs is the height of the waste package pier. Because of the modular design, the same waste package support can be used in both designs.

Further discussion on the various waste package designs can be found in Chapter 5 of this progress report.

Forecast: Development of waste package design information is an ongoing effort and the information needed and used in repository design will be updated as changes occur. These changes will be captured in either the MGDS Requirements Document or the Controlled Design Assumptions document as appropriate. This design activity will continue to report design information needed and provided for repository design as a result of the waste package design process.

4.1.6 Design Activity 1.11.3.1 - Area Needed Determination

The objective of this design activity is to determine the waste emplacement area required for the underground facility. Area needed is an important input to repository layout design and is also an important input in determining the suitability of the site for emplacing up to 70 000 MTU of high-level waste. The area needed is compared with the usable area (discussed in Section 4.1.7 of this progress report) to make this determination. Issues such as repository layouts and allowable waste package spacing are included in this activity.

Repository design work concentrated on optimizing the emplacement assumptions and arrangements to minimize the amount of excavation and the area required to emplace the 70 000 MTU inventory. Specific work focused on setting the areal mass loading at the highest feasible level, addressing the issue of the allocation of drift space for waste with low heat output, and re-examining the spacing between emplacement drifts. Analysis of areal mass loading and allocation of drift space are discussed below, while drift spacing is discussed in Section 4.1.17 of this progress report.

The areal mass loading assumed for the advanced conceptual design was 83 MTU/acre. A reassessment of this value began to set the viability assessment areal mass loading at the highest value that results in compliance with all thermal goals. Early indications are that the new thermal goal limiting peak temperature at the average top of the underlying zeolitic layer, discussed in

Section 4.1.16 of Progress Report #15 (DOE, 1997e), will represent the limiting condition for overall areal mass loading. Preliminary evaluations indicate that areal mass loadings of approximately 85 MTU/acre should result in meeting this goal. An increase in areal mass loading from 83 to 85 MTU/acre would slightly reduce the area required to emplace waste by about 20 acres, or just over 2 percent of the area requirement.

The waste inventory currently being used to guide design is described in Key Assumptions 003 and 004 of the Controlled Design Assumptions Document (CRWMS M&O, 1996c). This inventory shows a total of 10,938 waste packages, with 3,259 of these being defense high-level waste packages. These packages emit considerably less heat than the commercial spent nuclear fuel packages, but they have been allocated drift space commensurate with their heavy metal content "equivalency" in the advanced conceptual design. A different approach being evaluated is to place the defense high level waste packages in the empty spaces between the commercial spent nuclear fuel packages, allowing the defense high level waste packages no allocation of discrete drift space. This approach is also discussed in Section 4.1.17 of this progress report. If adopted, this approach would reduce the area required for the 70 000 MTU inventory by approximately 10 percent, or about 85 acres. The heat output of the defense high-level waste would be incorporated into thermal modeling, as would the heat output from the commercial spent nuclear fuel. This strategy would require accommodations regarding the high-level waste to allow the alternating placement of the commercial spent fuel and high-level waste packages. Such accommodations may include appropriate timing of the receipt of the high-level waste, surface or subsurface storage of commercial spent nuclear fuel or high level waste, or carrying one package over another in the emplacement drift.

Forecast: This design activity will be ongoing because its purpose is to update and refine repository layouts as additional information becomes available from design efforts and site characterization. An analysis defining the repository subsurface layout to be presented in support of the viability assessment is in progress and will be completed early in the next reporting period. (This analysis was forecast in Progress Report #15 to be completed this reporting period but was subsequently rescheduled to the next reporting period.)

4.1.7 Design Activity 1.11.3.2 - Usable Area and Flexibility Evaluation

The objectives of thi design activity are to

1. Analyze the three-dimensional structure and stratigraphy of Yucca Mountain to identify usable areas and ensure that sufficient area is characterized to allow design flexibility

2. Produce graphic cross sections and maps that can be used to lay out the drift arrangements

3. Cor. ..e drift arrangements to ensure that they fit the geology and structure

4. Identify site geologic data requirements for determining usable area.

This design act v ty is closely related to and builds on work performed under Design Activity 1.11.1.3, discussed in Section 4.1.3 of this progress report. This design activity also helps determine the suitability of the site for emplacing up to 70 000 MTU of radioactive waste.

During the last reporting period, initial development of a new three-dimensional computer model of the repository site area was completed under Design Activity 1.11.1.3 (see Section 4.1.3 of this progress report). The model was refined and updated this reporting period. The new model will provide additional confidence in Project assessment of available emplacement area. The usable area shown in this analysis is based on existing site characterization data or on data currently planned to be obtained. The usable area allows for expansion of the repository, if needed, from the current layout for 70 000 MTU at 83 MTU/acre

The new model and the determination of the usable area for repository siting is detailed in the Determination of Available Three Dimensional Volume for Repository Siting design analysis (CRWMS M&O, 1997f).

Forecast: Work will continue on extending the use of the LYNX model to define the available repository siting area. As the model is updated, the available area will also be modified.

4.1.8 Design Activity 1.11.3.3 - Vertical and Horizontal Emplacement Orientation Decision

The objective of this design activity is to provide the performance evaluation to document the decision on emplacement orientation. Well before this reporting period, the decision was made to proceed on the design assumption of horizontal in-drift emplacement. All subsequent work has involved waste packages emplaced horizontally in drifts. Work in this design activity addresses issues relevant to this emplacement orientation, such as retrievability and fea..ibility of placing backfill in drifts.

A report was completed that demonstrated the feasibility of placing backfill in emplacement drifts for the current planned emplacement mode (CRWMS M&O, 1997h). This report supports the Project's design goal of not precluding a future decision to use backfill should repository and waste package performance considerations warrant.

Forecast: Both repository and engineered barrier system designers will continue to evaluate the issues of waste package support mechanism and emplacement equipment design and methodology. A design analysis is being prepared that will describe the process of selecting the gantry-pedestal emplacement concept. This analysis should be approved in the next reporting period.

4.1.9 Design Activity 1.11.3.4 - Drainage and Moisture Control Plan

The objective of this design activity is to provide postclosure designs for the layout of the underground facility to limit the amount of water in contact with the container (emplaced waste package) and, by so doing, to provide a favorable containment and isolation environment. The objective is not only to limit the amount of water in contact with the waste packages, but also to promote the movement of moisture away from the waste emplacement areas.

The subject of water movement and drainage control was discussed in Progress Report #14 (DOE, 1996g). No new design work in this area has been performed since the release of the advanced conceptual design. The repository subsurface concept to be presented in the viability assessment incorporates the general drainage arrangement shown in the advanced conceptual design. The measures to control water movement include slight gradients in both emplacement drifts and mains to promote gravity flow of water away from waste packages. Minor changes in the main drift arrangement, though not directed toward moisture control, still adhere to the basic drainage concept.

Forecast: As subsequent phases of repository design progress, options for controlling moisture movement will be evaluated for their potential to improve repository performance.

4.1.10 Design Activity 1.11.3.5 - Criteria for Contingency Plan

The objective of this design activity is to provide criteria for the development of a contingency plan to deal with unexpected conditions that may be encountered during site characterization, repository construction, and the remainder of the preclosure period. Examples of unexpected conditions that may be encountered include small zones of perched water, localized heavy fracture zones, water recharge pathways, and localized heavily lithophysae-rich zones.

As discussed in Progress Report #15, the repository subsurface layout as it has evolved since the release of the advanced conceptual design continues to possess a high degree of flexibility to adjust to changes in concept as well as unexpected geologic conditions. No additional information is available this reporting period.

Forecast: Work on this activity will continue through FY 1997 and through progressive phases of repository design. The potential repository block definition will continue to be examined as information is received from site characterization activities. As more information on geologic structures is acquired, expansion and adjustment of the primary area will be considered.

4.1.11 Design Activity 1.11.4.1 - Chemical Changes Resulting From the Use of Construction Materials

The objective of this activity is to quantify the chemical changes (e.g., change in pH) that result from the use of a given quantity of construction material (e.g., cement).

Work has continued on the development of candidate concrete formulations to enhance the long-term postclosure geochemical performance of concrete. This effort has concentrated mainly on concrete formulations that act to minimize the potential pH of the concrete. Laboratory testing has been planned to examine specific chemical processes that affect pH at sustained high temperatures.

Forecast: Further study of corrosion processes and in situ thermal testing of potential ground support components are current tasks for FY 1997. Understanding of the degradation of materials during preclosure should enable an assessment of the impact on the long-term geochemical environment in emplacement drifts.

4.1.12 Design Activity 1.11.4.2 - Material Inventory Criteria

The objective of this design activity is to establish appropriate limits on the inventory of materials that will be used in construction and operation of the underground facility and to write criteria for the appropriate limits on the inventory of materials, including backfill, that will be left in the openings after decommissioning. Materials in the repository potentially affect geochemistry and therefore potentially affect waste package and repository performance. This design activity is closely related to and builds upon Design Activity 1.11.4.1, discussed in Section 4.1.11 of this progress report.

Preliminary qualitative guidelines for the usage of tracers, fluids, and materials for potential repository drifts were developed for use in design. Also, an initial description was prepared of the design information needed regarding possible uses of tracers, fluids, and materials. These data will be used in calculating chemical effects of tracers, fluids, and materials to evaluate potential impacts on waste isolation capabilities of the site.

Analyses have been performed to attempt to bound potential impacts to waste isolation from usage of tracers, fluids, and materials in the site characterization testing and ESF construction programs. However, the analyses performed for these activities generally depend on the specific tracers, fluids, and materials (or types of tracers, fluids, and materials) being evaluated, their mass, and their location relative to potential waste emplacement zones. In many instances, defining a limited quantity of a tracer, fluid, and material that can be left in the geologic system with the expectation of negligible impact to waste isolation is predicated upon the assumption that there is at least 37 m of tuff between the tracers, fluids, and materials deposition site and the closest potential waste package.

Because there is no mass of tuff between potential tracers, fluids, and materials deposition sites and waste emplacement zones in the repository drifts, even small amounts of substances represent large perturbations to the ambient system near potential waste packages. In addition, these perturbations could occur on a site-wide scale and not just as localized anomalies. Therefore, previous assessments of impacts cannot be usefully applied to the repository design and construction.

Qualitative guidelines, however, can be furnished for a number of general classes of tracers, fluids, and materials without additional analyses. In general, tracers can be used throughout the site with negligible impact to waste isolation expected because of the low concentrations used (~20 ppm), and in some instances the limited spatial extent of their use. This is particularly true for gas-phase tracers. One tracer that has been approved for use in subsurface construction water in all locations is LiBr (lithium bromide). The gas-phase tracer SF_6 (sulfur hexafluoride) has been approved for specific applications in the subsurface and in boreholes. Qualitative guidelines were developed for the following broad classes of fluids and materials that have been evaluated to date: water, organic-based fluids, concentrated salt solutions, solid salts (halides, nitrates, sulfates), alloys, earth materials, organic materials, and cementitious materials.

Of substances used for repository construction, operation or closure, only those that are left in the geologic environment postclosure (i.e., committed to the site, either intentionally or inadvertently) are of concern for impacts to repository performance, unless the substance alters the geologic environment before being removed from the site. To assess the potential effects to hydrology and geochemistry, and to evaluate the impacts to waste isolation resulting from the usage of tracers, fluids, and materials in a repository, the needed information consists of the following three parts: (1) the amount of the tracers, fluids, and materials to be committed to the site (i.e., emplaced and expected to be left after closure); (2) the distribution of tracers, fluids, and materials throughout the tunnels; and (3) the nature and composition of the tracers, fluids, and materials.

A computer data base is being developed to track construction materials used in the repository and other items that might become a part of the repository. Plans are for the database to contain seven categories of construction materials: permanently installed items, temporarily installed items, construction equipment, operating equipment, consumables, waste, and water. The data base will track quantity, type, location, system, and estimated service life and also give a description for each material category. The data base will also be able to account for items that are subsequently removed from the repository. The data base can be used by Project personnel to analyze the different materials used in repository construction or other items that become part of the repository.

Forecast: Work is in progress to address some of the needed constraints for potential repository construction. This activity will continue through the end of FY 1997. Any information generated relevant to constraints on tracers, fluids, and materials usage in potential repository construction will be used to update guidance for use of these materials and will be evaluated for inclusion in the materials data base. Work on the design and formulation of ground support materials, especially cementitious materials, will continue, and results will be used to develop material inventory criteria for the potential repository. An inventory of committed materials (materials to be left in the repository after closure) will be prepared. The data base for tracking construction materials used in the repository will be completed.

4.1.13 Design Activity 1.11.4.3 - Water Management Criteria

The objective of this design activity is to establish appropriate limits on the amount of water that will be used for surface and underground facility construction and operation.

No work has been performed for this activity during the reporting period. Related work is reported in Section 6.21 concerning water management for ESF construction and testing and surface-based testing activities.

The forecast for this section in Progress Report #15 predicted that criteria for water use in dust suppression, ground support installation, and fire protection would be input to the design that supports the viability assessment. This work was not performed because of priority conflicts and has not yet been rescheduled.

Forecast: No work is planned for this activity in the next reporting period. For related work, see Section 6.21 forecast.

4.1.14 Design Activity 1.11.5.1 - Excavation Methods Criteria

The objective of this design activity is to examine excavation methods available for repository construction and to identify constraints on those methods arising from postclosure performance considerations. The objective is to limit excavation-induced changes to rock mass permeability.

The majority of information presented on excavation methodologies in the MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b) and discussed in Progress Reports #14 and #15 is still current. The repository subsurface layout changes discussed in Progress Report #15 remain firm for the design to support the viability assessment. There will, however, be continuing refinements to simplify the layout and to minimize non tunnel boring machine excavation. The subsurface layout and excavation design analyses are currently in preparation.

Emplacement Drifts

A thermal management analysis in progress indicates that increasing emplacement drift spacing from 22.5 m to 28 m with correspondingly closer spacing of the waste packages in the emplacement drift may be viable. This arrangement maintains an areal loading density in the range of 80 to 100 MTU/acre for commercial spent nuclear fuel and still meets established thermal goals. The net effect would be to considerably reduce the number of emplacement drifts needed for 70 000 MTU of waste. The design of the waste package handling equipment that will operate in the emplacement drifts will define the final excavated drift diameter, which is currently set at 5.5 m. The emplacement drift entry openings are planned to be excavated by roadheader and the emplacement drifts by tunnel boring machine.

Central Exhaust Main and Ventilation Raises

The current design to support the viability assessment places the central exhaust main 10 m below the emplacement drifts. A short 10 m raise will connect each emplacement drift to the exhaust main. The main advantage of a short ventilation raise is the reduction in raise boring activity (less ground disturbance and lower cost) and shorter access ramps from the south and north mains to the exhaust main. The exhaust main will require continual ventilation to maintain a temperature level that will allow personnel to enter and periodically inspect and maintain the tunnel and equipment, such as monitoring equipment to detect radioactive materials in the exhaust air from the emplacement drifts.

Roadheader Excavation

The simplified design layout to support the viability assessment will seek to minimize roadheader excavation wherever possible. While developments in roadheader technology for hard rock excavation remain encouraging, this excavation method is less productive and more expensive than tunnel boring machine excavation for longer tunnel distances. The majority of roadheader excavation is expected to be for the turnouts connecting the mains and the emplacement drifts, and short connecting drifts where tunnel boring machine excavation would be impractical.

Performance Confirmation Drifts

The Subsurface Repository Performance Confirmation Facilities analysis (CRWMS M&O, 1997i) identified observation drifts across the repository block as a possible facilities solution to the requirements for collecting performance confirmation data. The drifts would be located 10 m above the emplacement horizon and excavated by either roadheader or by tunnel boring machine depending on the number required. Instrumented boreholes drilled from the observation drifts into the rock mass surrounding emplacement drifts would detect changes in rock characteristics resulting from the thermal effects of the waste packages.

Forecast: Activity on this task will continue through FY 1997 and through progressive phases of repository design. The layout concept is largely fixed for the design to support the viability assessment, although some refinement can be expected as the design analyses progress. Primary and secondary excavation methods will continue to be evaluated as the design evolves. The progress of prototypes of hardrock roadheaders will continue to be monitored to gauge their success and applicability to the Yucca Mountain Project.

The selection of the type and extent of facilities required for performance confirmation requires further study to determine the extent and frequency of performance confirmation data that must be collected. Such additional studies are not currently planned to occur until license application design begins.

4.1.15 Design Activity 1.11.5.2 - Long-Term Subsidence Control Strategy

The objective of this design activity is to evaluate the potential for postclosure surface subsidence and the impact of ground movement in the vicinity of the excavations on waste containment and isolation.

The potential for postclosure surface subsidence was described in Progress Report #14. Progress Report #15 stated that the Project has preliminarily concluded that subsidence is not a major issue and the subject will be revisited in the development of license application design to address the specific subsurface configuration at that time. As discussed in this section of Progress Report #15, subsidence is generally caused by pillar rather than drift failure. However, pillar failure begins at the drifts that border each pillar. Therefore, if the drifts are stable, the pillar is also. Thus, Project focus has been on drift stability rather than on subsidence. The effects of pillar stability on predicted surface subsidence were not addressed this reporting period.

Forecast: Evaluation of subsidence and drift stability will continue as part of an ongoing emplacement drift stability and maintenance task. Future analyses will concentrate on ground support and stability of individual drifts and will include use of a jointed rock mass model. Specific analyses to support subsidence predictions are not planned for the next reporting period.

4.1.16 Design Activity 1.11.6.1 - Thermal Loading for Underground Facility

The objective of this design activity is to establish the allowable thermal loading as a function of waste age and burnup, and waste package emplacement concept.

The ability to meet the overall performance requirements of the proposed repository at Yucca Mountain requires that the two major subsystems (natural and engineered barriers) positively contribute to the containment and isolation of radionuclides. In addition to the postclosure performance, the proposed repository must meet preclosure requirements of (a) safe operation and (b) maintenance of waste retrieval capability. The thermal, mechanical, hydrological, and chemical behavior of the repository underground facilities must be adequately understood to determine whether both the postclosure and preclosure requirements will be met in the presence of significant thermal loads.

Laboratory and in situ thermal testing and natural analog studies are key elements in gaining an understanding of how the Yucca Mountain rock would be affected by waste emplacement. The single-heater test has started, and the drift-scale test is under construction. These test programs are discussed further in Chapter 5 of this progress report.

Aside from continuation of the single-heater test and planning for the drift-scale test, no work on the thermal loading of the underground facility was conducted during this reporting period.

Forecast: The thermal loading strategy will continue to be refined as it is implemented in the engineering, site, and regulatory areas. The Thermal Loading Study for FY 1996 (CRWMS M&O, 1996o) results will be reconsidered if appropriate as the understanding of site characteristics evolves. As the results of in situ thermal tests, laboratory tests, and natural analog studies become available, they will be used to validate models to predict the thermal behavior of the natural and engineered systems. This process will allow a decision to be made before license application on the thermal loading of the potential repository.

4.1.17 Design Activity 1.11.6.2 - Borehole Spacing Strategy

The objective of this design activity as originally planned was to develop a strategy for spacing waste packages within the repository that included emplacement boreholes and drifts containing the emplacement boreholes. As stated in Progress Report #11 (DOE, 1995b), however, Project planning has changed from emplacement of waste packages in boreholes to in-drift emplacement of waste packages. Thus, the objective now is to develop a strategy for the spacing of emplacement drifts and waste packages within the drifts. Drift and waste package spacing have a major impact on temperatures in and around the waste packages and in the repository as a whole. These two parameters can be adjusted to obtain a desired repository thermal loading or to achieve design goals for temperature or humidity.

This activity is closely related to Design Activity 1.11.6.1, discussed in Section 4.1.16 of this progress report. The emphasis in Design Activity 1.11.6.1, however, is on developing an overall repository thermal load and related goals and constraints, while the emphasis in this activity is on developing the details to achieve the selected thermal load and meeting the established thermal goals and constraints. This activity is also related to the waste emplacement management concepts discussed in Sections 4.1.6 and 4.1.8 of this progress report.

As mentioned in Section 4.1.6 of this progress report, an effort is underway [see Repository Thermal Loading Management Analysis (CRWMS M&O, in prep.[a])] to re-evaluate the spacing between emplacement drifts. This spacing has remained at 22.5 meters since the Initial Summary Report for Repository/Waste Package Advanced Conceptual Design (CRWMS M&O, 1994a) was produced in August 1994. This spacing was included in the MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b) released in March 1996.

The combination of drift spacing, waste package spacing, and waste package mass content produces the areal mass loading value for the repository. The current evaluation is focused on maximizing the emplacement drift spacing consistent with compliance with thermal goals and with the incorporation of the concept for treatment of defense high-level waste packages discussed in Section 4.1.6. This concept would help minimize the amount of excavation needed to emplace waste. The analysis will set a drift spacing, for a given overall areal mass loading, by moving the drifts farther apart and the commercial spent nuclear fuel packages closer together within the drifts until the space left for the defense high level waste packages is just adequate to provide room for emplacing these packages. This configuration will then be modeled for thermal conditions to ascertain its compliance with the thermal goals. If all goals are met, the drift spacing will be considered valid.

Forecast: The thermal management design analysis containing the drift spacing work is scheduled for completion during the next reporting period. This analysis is expected to determine the most efficient method of spacing waste packages and emplacement drifts that meets requirements and limits.

4.1.18 Design Activity 1.11.6.3 - Sensitivity Studies

The objective of this design activity is to evaluate the effects of uncertainty in the description of the waste form and geologic setting. Recent activities have focused on determining predicted repository thermohydrologic and thermomechanical response to variations in thermal loading. This information will be used to evaluate the adequacy of data gathered and to determine with confidence whether repository performance goals have been met.

A Waste Isolation Study (CRWMS M&O, in prep.[b]) was conducted that documented the current understanding of performance of various components of the engineered and natural barrier systems. This study used information developed in the Description of Performance Allocation report (CRWMS M&O, 1996p) to identify the potential contributions these barriers may have in performing the overall waste isolation function. The study also examined the potential benefits of backfill in reducing the relative humidity at the waste package surface for the higher percolation fluxes currently being assumed by the Project for repository performance calculations and design purposes. Backfill calculations performed previously, using a less detailed near-field model and lower fluxes, showed an improvement in performance attributable to backfill of roughly one order of magnitude. The new calculations, with a more detailed near-field model and the higher fluxes, show no appreciable differences between the backfill and the no backfill options. Note, however, that the only backfill characteristics modeled in this analysis were its effects on relative humidity at the waste package surface of the waste package resulting from the insulating nature of the backfill. This impact on relative humidity then affected the time of onset and the rate of corrosion of the waste package but not enough to appreciably change performance. The study did not consider the capillary effects of backfill that may result in evaporation or diversion of water from the waste package or the potential value of the backfill in providing mechanical protection from rockfall.

This study also examined the potential bene ts of spent nuclear fuel cladding, waste package material galvanic protection, drip shields, and invert additives to long term waste isolation. Preliminary results indicate that very long-lived cladding, galvanic protection, and drip shields can provide significant performance. Completion of this study was delayed to the next reporting period to allow more thorough documentation of the findings of this report.

The Waste Quantity, Mix, and Throughput Study (CRWMS M&O, 1997j) was completed during this reporting period. This study provided an update of the waste quantities for all types of wastes that may be expected to be received on an annual basis at the repository. Potential extensions of these quantities were also identified. The study also identified the areas of the repository design - surface, subsurface, and waste package - that are sensitive to the variable waste stream parameters. Such sensitivities include the shipment rates, shipment cask types, thermal output of the fuel assemblies, etc. Design levels to be used for sizing surface and subsurface facilities were recommended. Finally, this study identified an approach and waste quantities to be used for sizing Lag Storage due to shipment surges, down time in the processing lines, or to accommodate the ability to manage the thermal output of the waste packages.

Forecast: The Seals Design requirements system study and the Waste Package system study are scheduled for completion during the next reporting period. The Waste Isolation Requirements System Study will also be completed at that time. In addition, a design basis concepts activity is planned that will continue to examine the sensitivities of the engineered barrier system and the natural barrier system to different parameter assumptions. This activity is expected to more fully examine the capillary benefits of backfill as well.

4.1.19 Design Activity 1.11.6.4 - Strategy for Containment Enhancement

The objective of this design activity is to document how design of the underground facility has taken into account containment of radionuclides with special emphasis on obtaining better performance of the waste package barrier by keeping it dry.

Since the last reporting period, the estimates of percolation flux at the repository horizon have increased. These increased fluxes could have an impact on the estimates of the performance of the engineered barrier components reported in Progress Report #15. The sensitivity of the performance effects of backfill to the higher fluxes is discussed in Section 4.1.18 of this report.

Forecast: A design basis concepts activity is planned that will continue to examine the effectiveness of the engineered barrier system and potential approaches for improving its performance.

4.1.20 Design Activity 1.11.6.5 - Reference Calculations

The objective of this design activity is to provide a set of calculations that documents predictions of postclosure thermal and thermomechanical response of the host rock. These calculations may be used to address performance assessment issues. Thermal and thermomechanical response analyses performed to satisfy this design activity can be divided into near-field and far-field analyses; the near-field analyses can be further divided into container-scale analyses and drift-scale analyses.

Current thermomechanical analyses are based on information derived from the ESF, especially rock modulus and thermal expansion data. Work continues on the development of as-built ESF information, including descriptions (location, extent, and type) of installed ground support elements and ground conditions encountered. The evaluation of geoengineering data includes comparing monitoring data on ground supports, the as-mapped rock classification, and the ground-support type. Parameters such as rock mass strength and modulus are derived from the as-mapped data.

Forecast: Near-field thermal and thermomechanical analysis of drift temperature and stability will continue as part of an ongoing task. In particular, jointed rock models will be used to examine the effects of long-term thermal and seismic load cases and to provide a basis for assessing postclosure drift behavior. The development of ESF as-built drawings and a baseline data base will continue during FY 1997.

4.1.21 Design Activity 1.11.7.1 - Reference Postclosure Repository Design

The objective of this design activity is to establish the information that will constitute the reference postclosure design for use in performance assessment and to document this information in the MGDS Advanced Conceptual Design Report (now complete) and in reports pertinent to the viability assessment and license application design.

The development continued of a list of information needed for performance assessment. The list of information needed, which first appeared in Progress Report #14, was modified slightly by adding the following items to it:

- Description of underground ventilation system
- Location of radioactive release sources
- Description of the waste package transporter and underground waste package handling operations.

Some of the needed information was made available for performance assessment.

Forecast: Design information needed for performance assessment will continue to be made available to support performance assessment as it is developed throughout FY 1997.

4.1.22 Design Activity 1.11.7.2 - Documentation of Compliance

The objective of this design activity is to document that the characteristics and configurations of the repository and engineered barrier systems have been adequately established to show compliance with both the postclosure design criteria of 10 CFR 60.133 and regulatory performance objectives. This compliance is to be demonstrated in documentation for license application design. This activity draws from the results of many other activities.

A complete status of this design activity was included in Progress Report #14 and updated in Progress Report #15. No updates to the demonstration of compliance resulted from work during this reporting period.

Forecast: Demonstration of compliance of the repository and engineered barrier designs with 10 CFR Part 60 will be refined as the designs are further developed in an ongoing process. This section of future progress reports will provide updates on the compliance demonstration as new reports and analyses become available.

4.2 REPOSITORY DESIGN CRITERIA FOR RADIOLOGICAL SAFETY (SCP 8.3.2.3)

Design Activity 2.7.1.1, discussed in Section 4.2.1 of this progress report, provides the status of work performed to evaluate repository safety design criteria and performance goals. Radiological safety analysis is an ongoing process throughout all design phases of the repository project.

The list of events identified previously for consideration as design basis events was evaluated. As a result, the 22 external events identified were grouped into 11 potential analysis groups. These groups were then prioritized on the basis of the potential impact of each on repository design, the availability of information to support each analysis, and the importance of having the analysis complete to support the viability assessment. Similarly, internal events have been grouped into 16 potential analysis groups to facilitate efficient evaluations of all potential design basis events.

In addition, pilot analyses are being performed for a nonmechanistic waste package failure in the subsurface and for selected surface events. The pilot analyses allow the development of a model for future consequence analyses, identification of potential issues, and the development of consistency within the various analyses to be produced within the multidisciplinary design basis event task team. The pilot analyses also allow other potential design basis events to be evaluated using any postulated releases and similarities to the pilot analyses.

One of the 11 specific external analyses in progress is the determination of the probability of an aircraft crash on the proposed repository surface facilities. This analysis will provide the basis for either screening out this external event or establishing it as a Category 1 or 2 design basis event as defined in 10 CFR Part 60. NUREG-0800 (NRC, 1987) has been selected to define the approach for making this determination.

4.2.1 Design Activity 2.7.1.1 - Design Evaluation for Compliance with Radiological Safety Design Criteria and Performance Goals

The objective of this design activity is to evaluate the repository design against the radiological safety design criteria and performance goals at each phase of the design and to provide feedback to the designers on needed corrections or modifications. This activity will result in a repository design that will protect repository workers and the public from radiological hazards during the preclosure period.

In Progress Report #15, the development of the preliminary MGDS hazards analysis (CRWMS M&O, 1996q) was discussed. This analysis identifies hazards with potential radiological consequences and is the basis for a systematic evaluation of hazards as potential design basis events. The hazards identified represent potential initiating events and design basis event scenarios for surface and subsurface repository operations. This analysis identified both external and internal events that required some degree of additional evaluation for potential impact on a repository facility. During this reporting period, each external event has been further evaluated (a) to permit grouping of events to allow evaluation of events in the most efficient manner and (b) to prioritize events based on potential impacts on a viability assessment of the repository design.

As a result, the 22 external events are currently grouped into 11 potential analyses. As the evaluation and design progresses, additional combinations of events may occur to recognize commonality between scenarios and to make efficient use of resources. The 11 external event analysis groups are as follows:

External Event Analysis Group	External Event(s)
1. Aircraft crash	Aircraft crash
2. Industrial/military activity Induced accident	Industrial activity Military activity
3. Loss of onsite/offsite power	Loss of offsite/onsite power Lightning Extreme weather fluctuations

External Event Analysis Group	External Event(s)
4. Seismic activity	Earthquake Subsurface fault displacement Surface fault displacement Static fracturing Debris avalanching
5. Winds/storms	Tornado Extreme wind Sandstorms
6. Rainstorm related	Flooding Landslides Debris avalanching due to rainstorm (including rain and snow loads)
7. Rockfall/steel set fall	Earthquake Subsurface fault displacement Surface fault displacement Subsurface static fracturing Dissolution
8. Fire hazards analysis	Fire (range) Lightning
9. Thermal/thermal cycling	Static fracturing
10. Safeguards and security	Inadvertent future intrusions Intentional future intrusions
11. Ashfall roof loading	Volcanism, ashfall

Sixty internal events were also identified in the preliminary MGDS hazards analysis by synthesizing subsurface and surface system functional areas of design and potential interactions that could impact a radiological waste form (e.g., disposal container, waste package, fuel assembly, radioactive waste processing system component). This list of events was generated by determining the applicability of internal generic events that could potentially interact with the waste form and result in a radiological release. Each of these events will require the following, either individually or in combination with other events:

- A frequency analysis that demonstrates the event is incredible
- An analysis that demonstrates a release does not occur as a result of the event

- A consequence analysis that demonstrates that the radiological consequences of the event are within regulatory requirements, or that identifies required preventative or mitigating structures, systems, and components to ensure the radiological consequences are within regulatory requirements.

- Some combination of the above.

Similar to the external events evaluations, internal groups have also been grouped to facilitate their evaluations. Potential surface design basis events are grouped into 11 evaluation groups as summarized below:

Surface Internal Event Analysis Group	Internal Event(s)
1. Shipping cask related	Shipping cask drop Shipping cask slapdown Shipping cask collision Shield door jams shipping cask Handling equipment drop onto shipping cask Truck/railcar derailment
2. Spent fuel assembly canister related	Spent fuel assembly canister drop Spent fuel assembly canister slapdown Spent fuel assembly canister drop onto sharp object Spent fuel assembly canister collision Spent fuel assembly canister drop onto disposal container System generated missile striking spent fuel Fuel damage due to welding process Fuel damage due to laser radiation
3. Spent fuel assembly related	Spent fuel assembly drop Spent fuel assembly drop onto sharp object Spent fuel assembly collision Spent fuel assembly drop onto disposal container System-generated missile striking spent fuel Fuel damage due to welding process Fuel damage due to laser radiation

Surface Internal Event Analysis Group	Internal Event(s)
4. High-level waste canister related	High-level waste canister drop High-level waste canister slapdown High-level waste canister drop onto sharp object High-level waste canister collision High-level waste canister drop onto disposal container System-generated missile striking high level waste container Fuel damage to welding process Fuel damage to laser radiation
5. Disposal container related	Disposal container drop Disposal container slapdown Disposal container drop onto sharp object Disposal container collision Spent fuel assembly canister drop onto disposal container Spent fuel assembly drop onto disposal container High-level waste container drop onto disposal container System-generated missile striking disposal container Decon system flooding Fuel damage due to welding process Fuel damage due to laser radiation Handling equipment drop onto disposal container
6. Waste package related	Waste package drop Waste package slapdown Waste package drop onto sharp object Waste package collision Equipment drop onto waste package Transporter derailment System generated missile striking waste package Fuel damage due to welding process Fuel damage due to laser radiation

Surface Internal Event Analysis Group	Internal Event(s)
7. Fire hazards related	Fire in waste handling building external to fuel handling area Fire in low level waste area Fire in fuel handling area
8. Liquid low-level waste related	Low-level waste drop Handling equipment drop onto low-level waste
9. Solid low level waste related	Low level waste drop Handling equipment drop onto low-level waste
10. Mixed waste related	Mixed waste drop Handling equipment drop onto mixed waste
11. Process-upset related	Thermal excursion/process upset in low-level waste system

Potential subsurface design basis events, which are grouped into four potential analyses, are summarized below:

Subsurface Internal Event Analysis Group	Internal Event(s)
1. In ramp/drift during transporting waste package into main drift	Transporter derailment in ramp or main drift Waste package car rolls out of transporter Runaway transporter/railcar Transport cask door jams waste package Rockfall onto transporter Steel set drop onto waste package Loss of waste package cart restraint in sloped emplacement drift Fire-hydrogen explosion (from batteries)
2. In drift during transfer of waste package at emplacement drift	Emplacement railcar derailment Waste package car rolls out of transporter Emplacement railcar collision with emplacement locomotive External unloading mechanism fails Transport cask internal off-loading mechanism fails

Subsurface Internal Event Analysis Group		Internal Event(s)
2.	In drift during transfer of waste package at emplacement drift (continued)	Transport cask door jams waste package Rockfall onto transporter Rockfall onto waste package/emplacement railcar Steel set drop on waste package Fire-hydrogen explosion (from batteries)
3.	In drift during emplacement of waste package into drift	Emplacement railcar derailment Waste package collision Runaway transporter/railcar Rockfall onto waste package/emplacement railcar Steel set drop onto waste package Loss of waste package cart restraint in sloped emplacement drift Fire-hydrogen explosion (from batteries) Thermal cycling of waste package
4.	In drift during preclosure	Steel set drop onto waste package Rockfall onto waste package/emplace... railcar Thermal cycling of waste package

The above internal events depend on the design and are based on the current revision of the preliminary MGDS hazards analysis. A revision is planned to update the hazards analysis to reflect recent design changes (as modified in the Controlled Design Assumptions Document).

Two sets of pilot analyses—one for surface and one for subsurface—are being performed to evaluate the environmental consequences of selected potential design basis events. The pilot analyses will establish the dose model and assumptions that will be used for subsequent radiological consequence analyses (i.e., for other events), and to serve as a basic template for these subsequent analyses. The surface pilot analyses include five cases. Two cases involve nonmechanistic waste package failure; one within the waste handling building and the other outside. Three drop event cases include one spent nuclear fuel assembly drop onto another assembly, drop onto the floor of one spent fuel canister containing 21 pressurized water reactor fuel assemblies, and one defense high-level waste canister drop onto another defense high-level waste canister. The subsurface pilot analyses address only one case, a non-mechanistic waste package failure within the subsurface facility.

These pilot analyses are the first to incorporate revised radiological safety criteria from the NRC's recently approved revision to 10 CFR Part 60 on design basis events. The design basis event 10 CFR Part 60 revision formally introduces the term "design basis event" and defines it on the basis of two categories of importance to radiological safety:

Category 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; and

Category 2 Other natural and man-induced events that are considered unlikely to occur before permanent closure of the geologic repository operations area, but sufficiently credible to warrant consideration, taking into account the potential for significant radiological impacts on public health and safety.

The two categories carry different dose consequence criteria. Because Category 1 design basis events are reasonably likely to occur at least once in the preclosure lifetime of the facility, the dose criteria correspond to the normal operating limits of 10 CFR Part 20. For the more unlikely Category 2, the dose criterion is 5 rem (total effective dose equivalent) at or beyond the nearest boundary of the preclosure controlled area. The categorization of design basis events may also change the dose model and assumptions that will ultimately be used for radiological consequence analysis.

An analysis is in progress to determine the probability of an aircraft crash on the proposed repository surface facilities. This analysis will provide the basis for either screening out this external event or establishing it as a Category 1 or 2 design basis event. NUREG-0800 is being used to define the approach for making this determination. The probability analysis will address all appropriate aspects of Section 3.5.1.6 of this NUREG: the requirements of 3.5.1.6.II.1 (a), (b), and (c) will be addressed first to determine if the event can be screened out without a specific evaluation. If the event does not meet all these requirements, the model defined in 3.5.1.6.III.2 will be implemented to determine an actual probability. A screening criterion of an event probability of less than 1×10^{-6} per year will be used as the basis for determining whether to eliminate this event as a design basis event. Data are being collected that will permit the above screening and any subsequent evaluations, if required. The evaluation is being prepared in accordance with QAP-3-9, Design Analysis.

Ongoing identification of waste package-specific design basis events to be used as input to the waste package design effort in support of the viability assessment is discussed in Section 5.1.3 of this progress report (Subactivity 1.10.2.3.5).

Forecast: The development of design basis event analyses in accordance with appropriate QA procedures will continue. Updates on all external and internal events will be provided. The aircraft hazards and ashfall analyses are expected to be completed and the hazards analysis revised.

4.3 NONRADIOLOGICAL HEALTH AND SAFETY (SCP SECTION 8.3.2.4)

This section describes work performed to address nonradiological health and safety issues and concerns. Discussions of general work concerning this topic follow. Design Activities 8.3.2.4.1.1 and 8.3.2.4.1.2, discussed respectively in Sections 4.3.1 and 4.3.2 of this progress report, provide the status of work performed to evaluate repository design against certain nonradiological safety design criteria and performance goals discussed in the SCP (DOE, 1988)

Because the safety and health of workers receives high priority in repository design, the objective of this design activity is to evaluate the repository design against the nonradiological safety design requirements, criteria, and performance goals at each phase of the design and to provide feedback to the designers on needed corrections or modifications.

System safety efforts for this period focused on radiological safety. Areas of nonradiological safety depend on new designs or off-the-shelf equipment that may be used in new applications, which may introduce new hazards. The design has not progressed sufficiently so that nonradiological safety determinations can be made.

Forecast: Nonradiological safety efforts will include reviews of the Preliminary System Safety Analysis (CRWMS M&O, 1996r) that was performed for the advanced conceptual design. These reviews will result in revisions or updates as required. The radiological scenarios will be screened and referred to the design basis events group, through the Preliminary Hazards Analysis, for resolution using design basis events process. Nonradiological scenarios in the Preliminary System Safety Analysis will be created or updated to reflect hazards scenarios for new design items and commercial off-the-shelf-items that are used in new applications and may pose new hazards. Typical use of commercial off-the-shelf-items will be reviewed but not included in the analysis unless the usage presents a new hazard. This is a proven approach similar to that used for job safety analyses that will be used as precedent.

4.3.1 Design Activity 8.3.2.4.1.1 - Design Activity to Verify Access and Drift Usability

The methods used to develop the conceptual design of the repository were based on preliminary data. The ESF offers an opportunity to verify these design activities and to substantiate or provide the basis for adjusting the design techniques used. The objective of this specific activity is to verify the design techniques used to develop the conceptual design of the repository and to determine the suitability of the ESF access ramps and drifts for repository use.

ESF excavation activities continue to demonstrate the viability of the tunnel boring machine for excavating the majority of the repository openings, and for roadheaders to excavate the balance of the openings. Drilling and blasting remains a back up method where mechanical methods may be overly difficult to execute. Discussions with equipment manufacturers and the Repository Design Consulting Board indicate that the current layout design and excavation methods selected are feasible and in line with current tunneling and mining practices.

The results of ESF activities have been used to develop data to confirm the ESF design and to provide a site characterization baseline for use by repository design and for performance confirmation. Key design parameters, such as rock mass modulus, rock mass strength, and joint friction and cohesion, which were originally derived from empirical relationships based on borehole data, have been recalculated based on data from tunnel mapping. The tunnel mapping includes scanline mapping and full peripheral mapping.

Correlations have also been made between predicted values of tunnel displacement and values from actual construction monitoring, such as tunnel convergence and displacement of ground support components (e.g., steel sets). Work continues on comparing the rock mass mechanical properties and the construction monitoring data with the as-built ground support to gain a better understanding of the controlling factors in order to develop a more site-specific approach to ground control.

Planning for a drift-scale thermal test to observe rock mass and ground support performance at elevated temperature has included the specification of measurement activities, which include: rock convergence, rock/lining convergence, rock mass strain, and concrete lining strain. Temperature measurements will also be made of the rock and the lining. Plans were made to monitor behavior during heating and to measure concrete mechanical properties before and after the test. These test results will be used to either validate or modify the current design studies.

Forecast: As repository design advances, repository data needs will become better identified, and design verification activities will be modified accordingly. Data gathered from the ESF design verification activities during ESF construction and testing will be used as input for developing the repository design during FY 1997 and future years and to substantiate assumptions identified in the Controlled Design Assumptions Document (CRWMS M&O, 1996c).

4.3.2 Design Activity 8.3.2.4.1.2 - Design Activity to Verify Air Quality and Ventilation

The objective of this activity is to assess the impact of site characteristics on the repository ventilation design to ensure a safe working environment (temperature, humidity, velocity and composition of air). Site characteristics will determine dust quantities produced during construction, in situ gas types and quantities, and the wall roughness required for ventilation flow calculations. Data gathered from construction monitoring and site characterization will be used as input for repository ventilation design.

Field ventilation data (flow rates and pressures) were collected for the current ESF ventilation system mode of operation. The airway resistance factor for the ESF ventilation duct, which was established from actual ventilation measurements, is being used in the ventilation analysis for the repository subsurface design to support the viability assessment.

Additional data on dust concentrations were obtained from ongoing monitoring of the ventilation system supporting the current tunneling operations. The field dust information is being evaluated and considered in the development of the repository subsurface ventilation arrangement.

Forecast: A ventilation analysis is being developed as a part of repository subsurface design to support the viability assessment. Field ventilation measurements for verification of the airway resistance factor for the ESF tunnel will be conducted after the completion of the tunnel boring machine excavation of the south ramp.

4.4 PRECLOSURE DESIGN AND TECHNICAL FEASIBILITY (SCP SECTION 8.3.2.5)

4.4.1 Design Activity 4.4.3.1 - Operations Plan to Accompany the Advanced Conceptual Design

The objective of this design activity is to produce an operations plan to accompany the advanced conceptual design. A plan is needed to effectively design and evaluate the preclosure performance of the potential repository.

As reported in Progress Report #14, the advanced conceptual design has been completed, and therefore, this activity has been closed.

Forecast: This activity is closed.

4.4.2 Design Activity 4.4.3.2 - Operations Plan to Accompany the License Application Design

The objective of this design activity is to produce an operations plan to accompany the license application design. A plan is needed to effectively design and evaluate the preclosure performance of the potential repository.

Under the current plans and schedules, license application design is not scheduled to begin until FY 1998. Because advanced conceptual design is complete, work performed before FY 1998 in developing a repository operations plan for viability assessment will be reported under this design activity.

The update to the MGDS Concept of Operations (CRWMS M&O, 1996s) was reported in the previous period; no further update has occurred. The following summarizes the primary design analysis and technical document development activities that will impact repository operations. Some of these are ongoing and will be reflected in future releases of the concept of operations and the system design description documents.

Repository Surface Design

Waste Handling Systems Configuration Analysis. This analysis (CRWMS M&O, 1997d) was performed to determine the design concept required for the waste handling building operations in response to a change from predominantly canistered to uncanistered waste. The analysis compared wet and dry handling concepts, preliminarily selected a wet handling approach, and modeled the concepts using the computer simulation program WITNESS. Handling operations were selected and modeled for the shipping casks and canisters, bare and canistered fuel, and disposal container operations. The impact of equipment reliability and use of the operating stations was determined. The analysis also provided time and motion input to the operations and staffing analysis, discussed below.

Operations and Staffing Analysis. This analysis (CRWMS M&O, 1996t) developed a preliminary operating philosophy for the repository, including management and control, maintenance and supply, shift operations, and emergency response. The advanced conceptual design operating philosophy and staffing estimates were updated considering changes in the designs. The analysis also incorporated new direction associated with uncanistered fuel handling, shipping cask maintenance concepts, fabrication of emplacement drift structures at the site, and operating philosophies defined in the analysis. The operational concepts and staffing estimates were influenced primarily by a centralized philosophy for management and maintenance and by the design change to support uncanistered waste.

Radiological Safety Design Analysis. This safety analysis will use the MicroShield V4.2 (CRWMS M&O, 1997k) gamma shielding software to analyze gamma shielding and to estimate the personnel exposures from radiation. Required shield wall, doors and barrier thickness will be determined in each candidate area of waste handling building operations, and determinations will be made of acceptable levels of dose rates. The software was qualified during this reporting period, and preliminary assessments are in progress.

Waste Handling Operations Dose Assessments. This analysis, for which planning is in progress, will determine occupational dose rates for operating personnel, assess the effectiveness of the shielding, and re-assess the shielding approach. Time and distance information will be provided from the design activities, and may be supported by the WITNESS simulation model.

Repository Subsurface Design

Thermal Management Analysis. The thermal management analysis in progress (CRWMS M&O, in prep.[a]) is considering further refinement of waste emplacement and development design approaches. The spacing between waste packages in the emplacement drifts and the spacing between emplacement drifts are being re-evaluated, as discussed in Section 4.1.17 of this progress report. An increase in the drift spacing with a corresponding reduction in waste package spacing should minimize the number of emplacement drifts that need to be excavated and have no discernable impact on repository operations.

Retrieval Design Analysis. This analysis (CRWMS M&O, in prep.[c]) is identifying and providing concepts for the equipment required to retrieve a waste package from an emplacement drift. Equipment failure modes are being assessed, and the basic activities and sequences required for recovery operations are being determined.

Performance Confirmation Data Acquisition System. A design analysis (CRWMS M&O, 1997m) demonstrated that it is feasible to use a remote controlled gantry to travel through an emplacement drift and obtain data for performance confirmation without first having to cool the emplacement drift. Such a system would result in acquisition of better data and also less impact on the emplacement drift and waste packages.

Emplacement Design Analysis. An analysis (CRWMS M&O, in prep.[d]) in the final stages of review is considering various options to replace the advanced conceptual design railcar approach for emplacing the waste package in the emplacement drifts. The analysis concluded that using a gantry is the preferred option. This approach reduces the number of personnel required for operation, uses less equipment, and will likely be more reliable and safer. The two track transporter design used in the MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b) was replaced with a single track design. The design is expected to minimize subsurface maintenance operations.

Systems Engineering

Systems Design Descriptions. The system summaries portion of the system design description documents were completed and a draft of the Writers Guide was developed during this period. The operations sections of each system design description document will require that the operational philosophy for each system be described, including outlining organization and staffing requirements, shift operations, interface procedures, and operating and maintenance procedures.

Forecast: The design effort to support the viability assessment will continue, and additional operations concepts will be developed. The overall MGDS operating concept, as well as alternative concepts, will be captured in the next update of the Preliminary MGDS Concept of Operations Document. The reference design will be captured and documented in a new document titled Reference Design Description Document. The waste handling building operations simulation will be expanded as the design develops to accurately assess system performance, outline operating time lines, determine manpower requirements, and support the dose assessment activities. Writing of portions of the system design description documents other than the summary portions will begin during the next reporting period. However, the preliminary operational sections may not be addressed until license application design begins.

4.4.3 Design Activity 4.4.4.1 - Repository Design Requirements for License Application Design

The objective of this design activity is to develop repository design requirements for license application design.

The Repository Design Requirements Document (DOE, 1994f) captured the initial set of design requirements to support the development of license application design. As a result of streamlining the document hierarchy, repository design requirements will be captured at the

Project Level in a revision of the MGDS Requirements Document (DOE,1996b). Revision 3 of the MGDS Requirements Document has been drafted and is in review within the Civilian Radioactive Waste Management System Management and Operating Contractor. Once Revision 3 is approved, the Repository Design Requirements Document will be removed from Level 2 Change Control Board control. The status of the revision to the MGDS Requirements Document, as well as the status of the Repository Design Requirements Document, is provided in Appendix B of this progress report.

Forecast: The updated MGDS Requirements Document will be submitted to DOE and reviewed and approved by the Change Control Board, which is discussed in Section 2.1.3 of this progress report. The MGDS Requirements Document will then be maintained as required.

4.5 SEAL CHARACTERISTICS (SCP SECTION 8.3.3.2)

4.5.1 Study 1.12.2.1 - Seal Material Properties Development

The ability to seal openings to and inside a repository could significantly impact postclosure repository performance. The Yucca Mountain sealing program concentrates on cementitious and earthen materials emplaced in shafts, ramps, and boreholes. The strategy for sealing the proposed repository is to place seals in the shafts, ramps, and boreholes so that they do not act as potential pathways for water and gas flow. Current efforts are focused on in situ and laboratory testing and analysis of cementitious seal components planned for use in sealing exploratory boreholes at Yucca Mountain. Potential sealing locations include nonwelded Topopah Spring Tuff and the Calico Hills Formation.

An issue of importance for sealing is the longevity and durability of emplaced seals, particularly seal materials. Seal performance for all potential seal locations is related to initial performance (including the effect of emplacement techniques), as well as to long-term performance (mechanical/hydrologic/geochemical stability).

Activity 1.12.2.1.1- Detailed property determination of cementitious-based and earthen materials. Activity 1.12.2.1.1 concerns the determination of detailed properties of cementitious and earthen materials. The objective of this activity is to conduct laboratory testing and analysis to determine material properties for seals.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: Future testing and analysis will continue to focus on evaluating the performance of cementitious seals emplaced in thick-walled cylinders of tuff. Additional durability tests conducted under accelerated and extreme environmental conditions are also planned. Resumption of this work awaits funding. The schedule and need for future work in this area will depend on the results of the seals systems study to be completed in FY 1997. (See the forecast for Section 4.5.5 of this progress report.)

<u>Activity 1.12.2.1.2 - Hydraulic conductivity and consolidation testing of crushed tuff</u>. This activity concerns establishing the hydraulic conductivity and consolidation behavior of crushed tuff to support the development of criteria for shaft fill and drift backfill.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No activity is planned for FY 1997. The schedule and need for future work in this area to support decisions on sealing and backfill will depend on the results of the seals systems study to be completed in FY 1997. (See the forecast for Section 4.5.5 of this progress report.)

4.5.2 Design Activity 1.12.2.2 - A Degradation Model for Cementitious Materials Emplaced in a Tuffaceous Environment

The objective of this activity is to develop a degradation model that will provide insight into how material properties of sealing components, especially permeability and strength, could be altered after being in contact with tuff.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No activity is planned for FY 1997. The schedule and need for future work in this area to support decisions on sealing will depend on the results of the seals systems study to be completed in FY 1997. (See the forecast for Section 4.5.5 of this progress report.)

4.5.3 Study 1.12.2.3 - In Situ Testing of Seal Components

The objective of this study is to conduct in situ testing and analysis to evaluate the behavior of selected sealing components under both realistic in situ conditions and under unlikely conditions.

No progress was made during the reporting period, this was an unfunded activity.

Forecast: Resumption of work on planning for in situ small-scale borehole tests will depend on the results of the seals systems study to be completed in FY 1997. (See the forecast for Section 4.5.5 of this progress report.)

4.5.4 Design Activity 1.12.4.1 - Development of the Advanced Conceptual Design for Sealing

The objective of this activity is to provide the conceptual seal design for the repository. However, as reported in Progress Report #14, advanced conceptual design has been completed, and therefore, this activity has been closed, and all further work will be reported under Design Activity 1.12.4.2.

Forecast: This activity is closed.

4.5.5 Design Activity 1.12.4.2 - Development of the License Application Design for Sealing

Design Subactivity 1.12.4.2.1 - Define subsystem design requirements. The objective of this design subactivity is to refine design requirements that will assist in the development of sealing components for the license application design.

A systems study to address requirements for sealing the repository shafts, ramps, and boreholes began. With regard to boreholes, this study is addressing the need for sealing boreholes resulting from surface-based testing or subsurface testing. Specifically, the study is developing requirements for seals, and is determining the role seals must play in establishing compliance with 10 CFR Part 60 with respect to waste isolation and to returning the site back to its natural condition. The study also will determine the approach for meeting State and county regulations related to abandoned boreholes and the function the seals must play in meeting those requirements. This study is evaluating earlier work performed on seals before the development of total system performance assessment, before a great deal of site data were collected, and before the advanced conceptual design was completed. The earlier results will be updated, if necessary, to support the seal design activities.

Forecast: The systems study is expected to be completed during the next reporting period.

Design Subactivity 1.12.4.2.2 - Perform trade-off studies to support license application design development. The objective of this design subactivity is to provide technical justification for the selection of the final seal designs.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: This activity will be addressed in time to support license application design.

Design Subactivity 1.12.4.2.3 - Develop license application design for seals. The objective of this design subactivity is to provide the license application design for seals.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: This activity will be addressed in time to support license application design.

CHAPTER 5 - WASTE PACKAGE AND NEAR-FIELD ENVIRONMENT

INTRODUCTION

The waste package consists of the waste form (spent nuclear fuel assemblies) or canistered waste form (canistered fuel or canisters of defense high-level waste glass), fill gas, and a disposal container. The fuel canister is an outgrowth of the former multi-purpose canister effort, and refers to canisters intended primarily to facilitate handling of the waste. Per 10 CFR 60.2, the waste package may also include shielding, packing, and other absorbent materials, but these materials have not been implemented in current designs.

The introductory discussion in Chapter 4 of this progress report, which describes the current focus of repository and waste package design, is applicable to waste package design but is not repeated in Chapter 5.

The waste package program includes (a) the development of waste package design bases, (b) development of a reference design, (c) analysis of design, testing, and modeling of container materials, (d) testing and modeling of the waste form, and (e) working with other Yucca Mountain Site Characterization Project (Project) elements to characterize the waste package environment. This chapter describes progress in waste package design and progress in the testing and modeling activities that support that design. For convenience, it also discusses the waste package supports and inverts, although these configuration items are not part of the waste package itself. In addition, consistent with the Site Characterization Plan (SCP) organization, this chapter discusses the work that is characterizing the predicted relationship between the waste package and its environment. Waste package materials testing and modeling in support of performance assessment are described in Chapter 6.

Waste package design is an integrated program effort to develop a waste package that will meet performance objectives and design criteria for the potential repository. Among the more significant objectives are the following:

1. Waste package lifetime should be well in excess of 1000 years.

2. The waste package container must contrib to controlling the release rate of radionuclides during the period of isolation.

3. Criticality must be controlled during the period of regulatory concern.

The relationship between a favorable viability assessment and achievement of these objectives was discussed in Progress Report #14 (DOE, 1996g) and the Controlled Design Assumptions Document (CRWMS M&O, 1996c). The waste package is also expected to contribute significantly to meeting the interim performance standard as documented in the Controlled Design Assumptions Document Key Assumption 060

The following discussion summarizes some of the more notable waste package and engineered barrier system activities that occurred during this reporting period. More details are given in Sections 5.1 through 5.4.

In reading the following text recognize that the Project's understanding of site characteristics continues to evolve. For example, there are several indications that percolation flux may be considerably higher than was previously thought to be the case. Some of the studies and analyses discussed in this chapter assumed a lower percolation flux than the range of values supported by current information and analysis results. Therefore, the results of these studies will be re-evaluated as appropriate to address the evolving understanding of site characteristics.

Evolution of Waste Package Design

As systems strategies and long-term performance strategies have evolved, waste package designs have also evolved. As reported in Progress Report #15 (DOE, 1997e), disposal container design is now focused on the following four basic types (as discussed in Section 5.1.1), all of which are robust, multibarrier, metallic designs:

1. Disposal container for uncanistered fuel assemblies

2. Disposal container for canistered fuel

3. Disposal container for defense high-level waste glass pour canisters

4. Disposal container for defense high-level waste glass pour canisters and U.S. Department of Energy (DOE)-owned spent fuel.

Two disposal container designs for uncanistered fuel are being analyzed, one with a capacity of 21 pressurized water reactor fuel assemblies and another with a capacity of 44 boiling water reactor fuel assemblies. Additional designs with smaller capacities may be necessary for fuel that is unusually reactive or has an unusually high heat output. The Project baseline has been extended to encompass DOE-owned spent nuclear fuel; a modified version of the disposal container for defense high-level waste glass is also being considered. The modified design is intended to accommodate DOE-owned spent fuel along with waste glass canisters. DOE-owned spent fuel includes defense and research reactor fuel, as well as fuel from certain demonstration reactors (e.g., Shippingport and Fort St. Vrain). The sizes and capacities of the disposal container designs for canistered fuel will be chosen to accommodate the canisters to be placed in them.

As reported in Progress Report #15, emphasis has been shifted away from canistered commercial fuel because of the cancellation of the multi-purpose canister effort. Work during the reporting period, therefore, continued to focus on the other types of containers. Unless there is renewed emphasis on canisters for commercial fuel, the disposal container for canistered commercial fuel will remain at the advanced conceptual design stage (i.e., with little or no further design development). Because the current planning assumption is that Navy spent fuel will be emplaced in canisters, future work on development of a waste package for such canisters is anticipated.

Figures 5-1 through 5-4 illustrate disposal container designs under consideration. These designs are intended for pressurized water and boiling water reactor spent nuclear fuel, defense high-level waste glass, and DOE-owned spent fuel.

The container design shown in Figure 5-1 would hold 21 uncanistered pressurized water fuel assemblies. In this design, the neutron-absorbing material is in the form of interlocking plates, which do not require welding. Carbon steel tubes contribute to conducting heat to the surface of the waste package away from the fuel. However, a different design with aluminum thermal shunts in the fuel basket may still be necessary for fuel with an unusually high heat output.

For the design shown in Figure 5-2, (boiling water reactor waste container), the grid openings in the basket would be smaller than for the design shown in Figure 5-1 because of the smaller cross section of a boiling water reactor assembly, and because of their lower heat output, boiling water reactor assemblies do not require basket tubes or aluminum thermal shunts.

Two designs for defense waste glass are under consideration. The first, shown in Figure 5-3, would contain four waste glass canisters. The second design shown in Figure 5-4, would hold both waste glass and DOE owned spent fuel. This second design provides for five high-level waste glass canisters surrounding a center gap proposed for disposal of DOE-owned spent nuclear fuel. A circular tube would be placed at the center of the disposal container, and a basket would be installed in the tube. The basket cells would hold DOE-owned spent nuclear fuel. The basket design may vary among waste packages, depending on the type of DOE spent fuel to be disposed of in a given waste package.

Materials Selection

Work during this reporting period emphasized materials for the waste package supports and the invert materials rather than the waste packages themselves. This emphasis will continue into the next reporting period and the results of materials selection for these components will be reported in the next progress report.

With regard to waste package materials, current proposed materials are as follows:

- ASTM A 516 (carbon steel) and ASTM B 443 (Alloy 625) remain the materials selected for the corrosion-allowance and corrosion-resistant barriers, respectively. These material selections are unchanged since the last reporting period, but may change pending negative results from corrosion testing. ASTM A 516 remains the reference material for the basket tubes and basket guides.

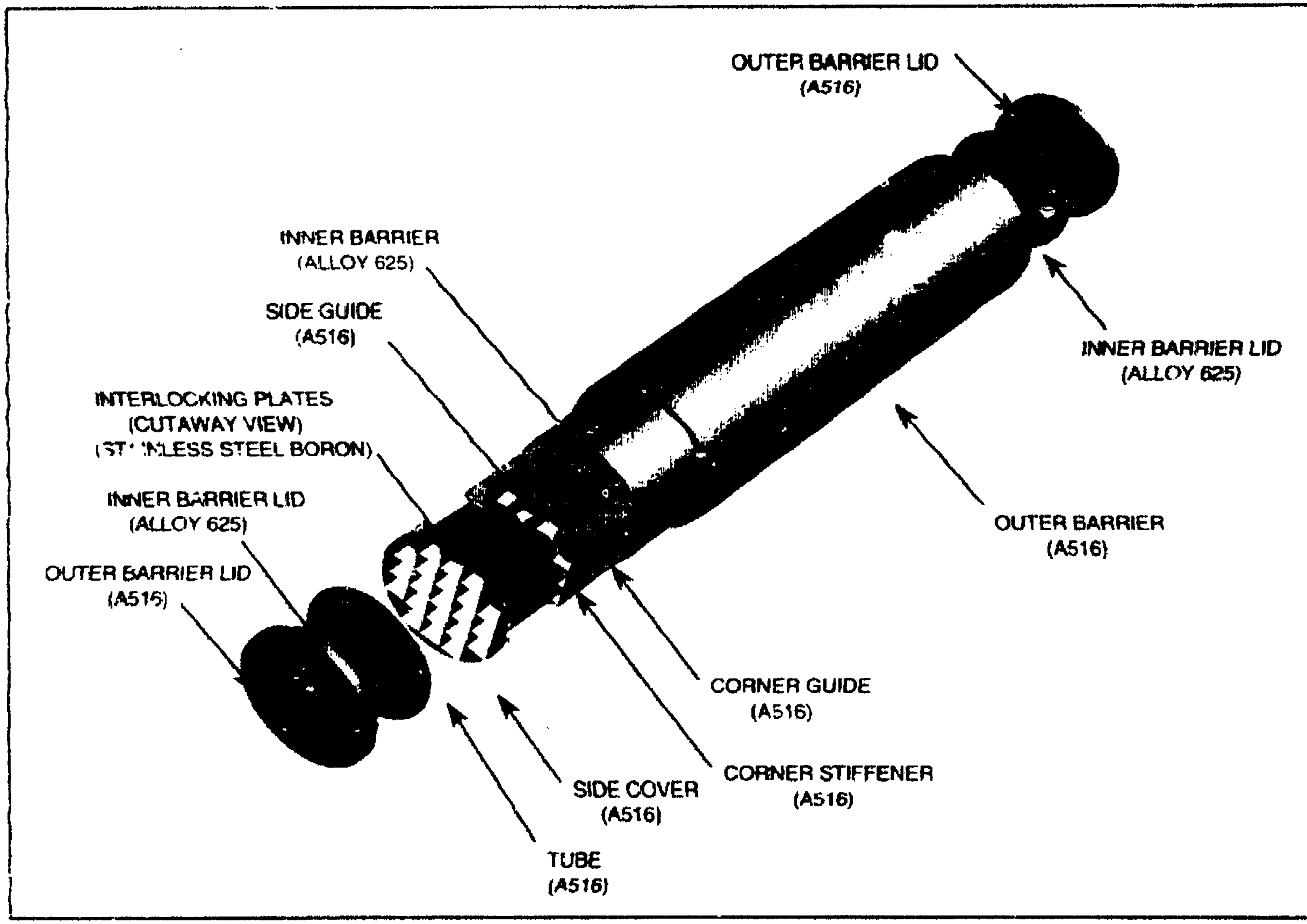

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Figure 5-1. Schematic of Disposal Container for Uncanistered Pressurized Water Reactor Spent Nuclear Fuel

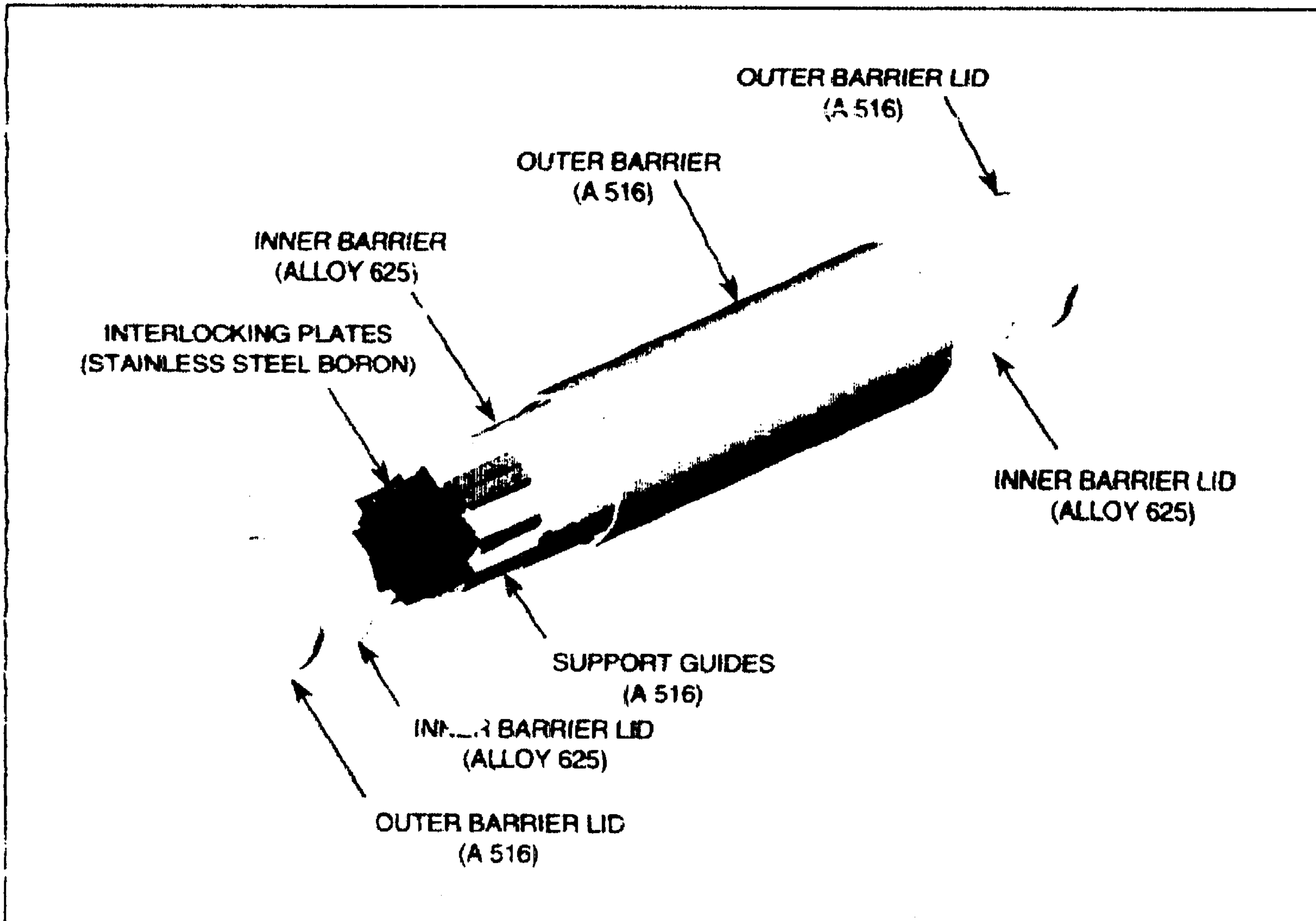

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Figure 5-2. Schematic of Disposal Container for Uncanistered Boiling Water Reactor Spent Nuclear Fuel

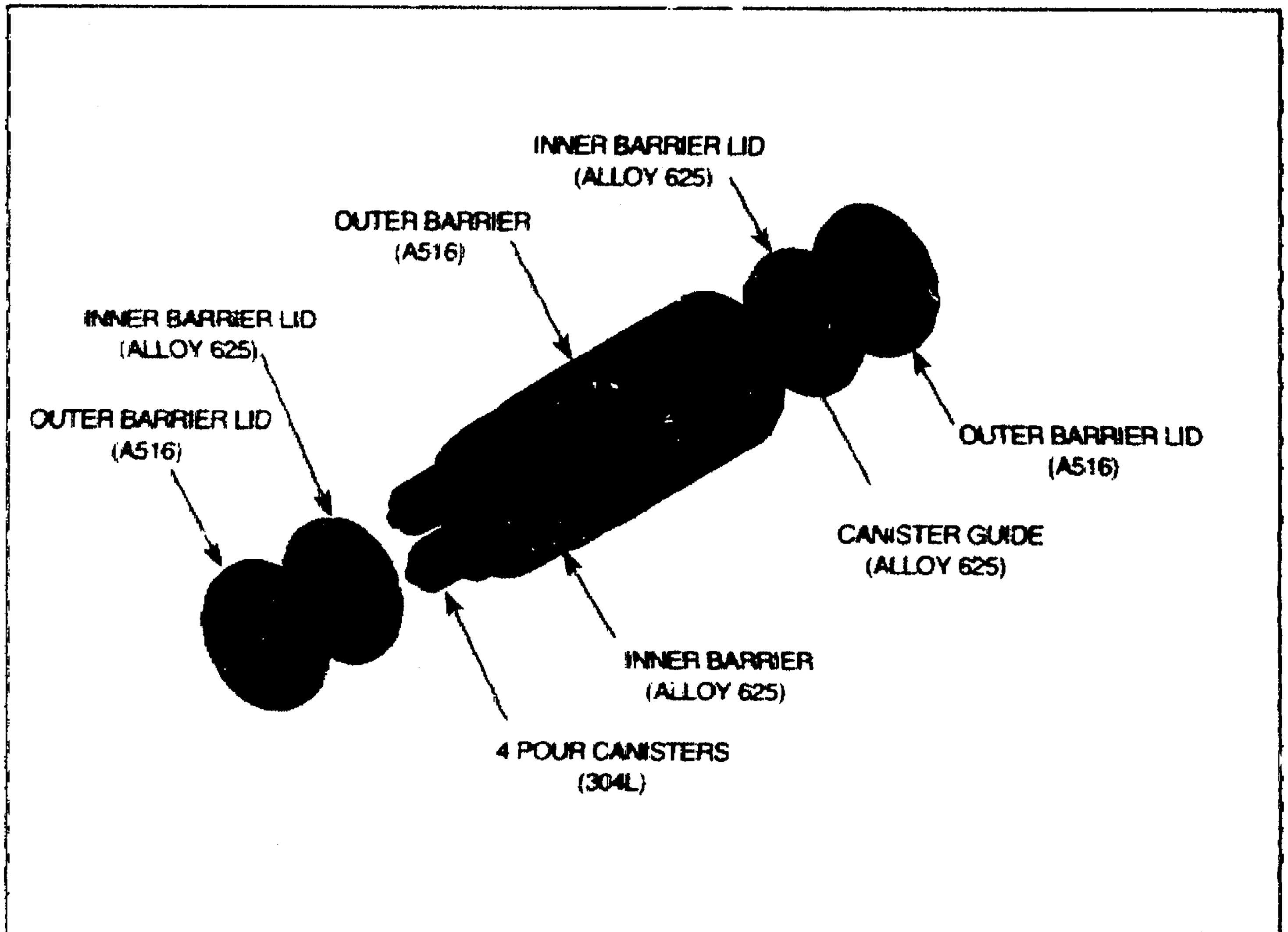

Figure 5-3. Schematic of Disposal Container for Defense High-Level Waste Glass

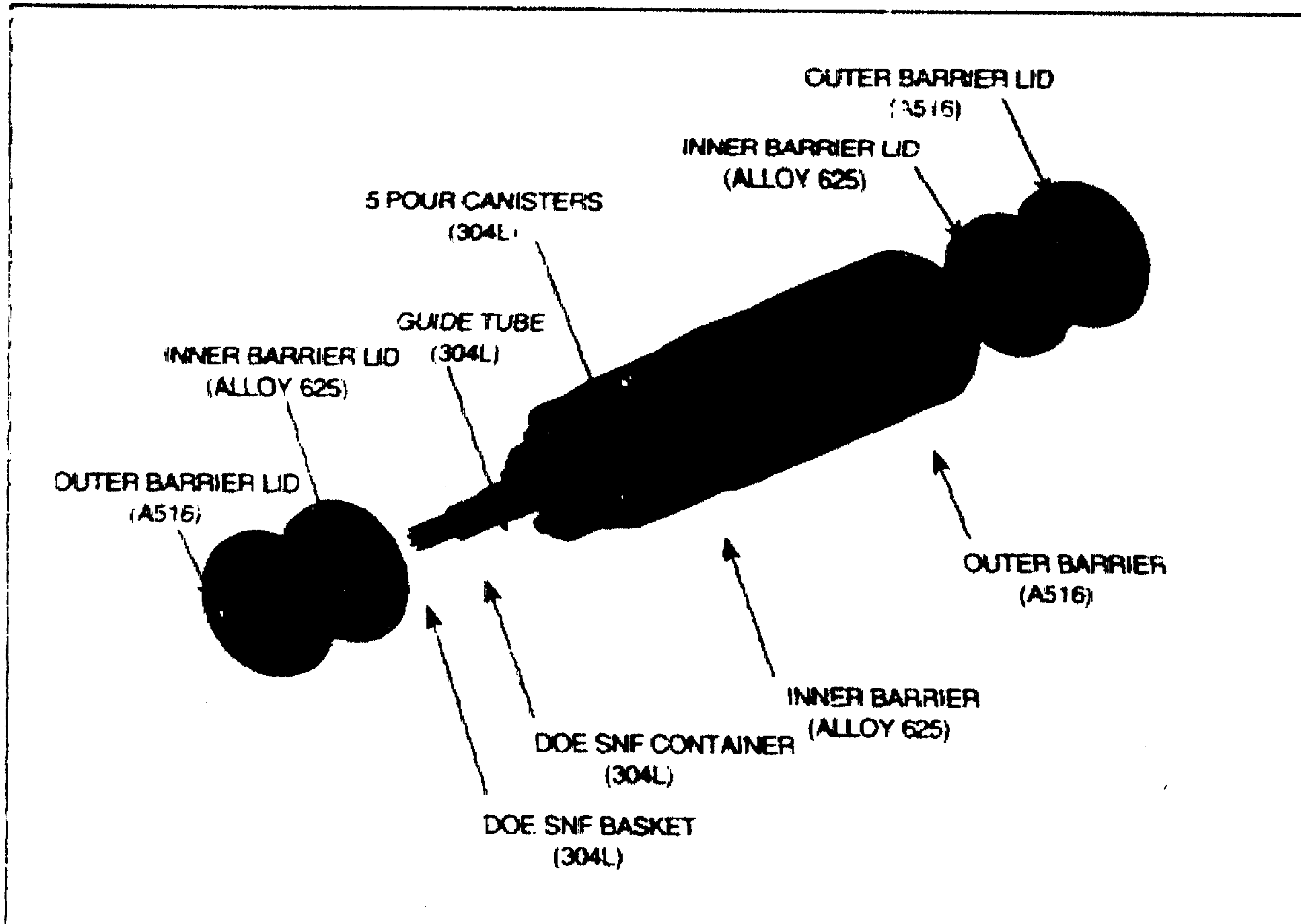

Figure 5-4 Schematic of Disposal Container for Defesne High-Level Waste Glass and DOE-Owned Spent Nuclear Fuel

- Neutronit A978 or equivalent remains the reference material for spent nuclear fuel basket criticality control primarily because of its superior corrosion resistance.

- Helium remains the reference fill gas.

Because Project emphasis remains on uncanistered fuel, no additional work on filler material is planned. See Section 5.1.4 for additional information on materials selection and analysis.

Design Analyses

Design analyses, discussed in Section 5.1.2, continue to focus on thermal, structural, and criticality analyses.

Thermal design efforts have advanced in three main areas: (1) evaluating the repository and emplacement drift thermal behavior and its impact upon waste packages, (2) evaluating waste package thermal conditions with regard to meeting the licensing requirements, and (3) evaluating and designing the waste package support and invert on the basis of its thermal and structural performance under nominal Mined Geologic Disposal System (MGDS) repository conditions.

Two major activities were included reporting period. One activity focused on the preliminary design of the waste package sup and pier. A design analysis report, entitled Waste Package Support and Pier Static and Seismic Analyses (CRWMS M&O, 1997n), determined appropriate dimensions and materials for the waste package support and pier using structural requirements.

The second activity focused on the waste package structural analyses. The Drop Analyses of Uncanistered Fuel Waste Package Designs (CRWMS M&O, in prep.[[e]) is currently in progress. The purpose of this analysis is to determine component dimensions. The component dimensions are required to show the adequacy of the uncanistered fuel waste package design with stainless steel boron neutron absorber plates under loading encountered during waste package drop events.

Criticality activities performed continued to focus on resolving disposal (postclosure) criticality issues. A meeting was held in February 1997 with the U.S. Nuclear Regulatory Commission (NRC) to discuss NRC staff comments on the Disposal Criticality Analysis Methodology Technical Report (CRWMS M&O, 1996d). The revision of the report will be completed next reporting period.

Production Technologies

Estimates for material and manufacturing continue to be collected from various manufacturers and vendors. This is an ongoing process.

Nevada Line Procedure-7-3 which describes how the programs to develop waste package closure processes will be conducted, was developed and approved. The Waste Package Closure Methods Technical Guidelines Document (CRWMS M&O, 1997o) was written, and development work started in March 1997.

In parallel with the closure weld activity, inspection techniques for closure welds are being developed. The Nondestructive Examination Technical Guidelines Document has been written (CRWMS M&O, 1997p), and the development program began in March 1997.

See Section 5.4 for additional information on waste package production technologies.

Near-Field Studies

Understanding of near-field characteristics and behavior is essential to accurately predict waste package and repository postclosure performance. During this reporting period, the Project continued to improve its understanding of near-field thermal-hydrological-mechanical-chemical processes. In addition, laboratory and design work in support of important in situ thermal tests continued. Major accomplishments included the following:

- A parameter sensitivity study used drift-seepage model to investigate the relationship between water seepage into the drift (drift seepage flux) and percolation flux for both homogeneous and heterogeneous conditions. The modeling indicated the key parameters affecting the predicted threshold percolation flux for seepage into the drift were the distribution of aperture sizes for the fractures and the heterogeneity of the major fracture flow paths. A relatively narrow distribution of aperture sizes reduces the threshold percolation flux at which water is predicted to be able to seep into the drift. The results of this work will be used to develop a more physically-based, predictive capability for determining the fraction of percolation flux that can enter a drift (see Section 5.2.3).

- Pre-test analyses of the drift-scale thermal test in the Exploratory Studies Facility (ESF) were conducted with three-dimensional NUFT-based models to determine (a) the maximum expected temperature rise at selected locations in the thermal test area, (b) the ventilation requirements in the neighboring drifts, and (c) the insulation requirements for the thermal bulkhead that separates the heated and unheated portions of the heater drift. This information was provided to the ESF test and design organizations for use in the design and construction of the drift-scale test (see Section 5.2.3).

- A sensitivity study of the influence of percolation flux on temperatures in the drift scale test was conducted with a two-dimensional thermal-hydrological model of the drift-scale thermal test. For the 5-mm/yr case, the maximum predicted drift-wall temperature at the center of the heater drift was more than 100°C lower than for the 0.05-mm/yr case. The 5-mm/yr case created a vertical dryout zone, but the zone was only one half as thick as in the 0.05-mm/yr case (see Section 5.2.3). The results of this study indicate that temperature and dryout measured on the drift scale test may be strongly indicative of percolation flux, even at percolation fluxes higher than 5 mm/yr.

- The single-heater test was modeled using a three-dimensional thermal-hydrological NUFT-based model that represented the effect of heat and mass transfer with all three ventilated drifts surrounding the test area. The model was used for a sensitivity study of bulk permeability. Results indicated the dryout zone volume increases with increasing bulk permeability, while the temperatures inside the boiling and superheated zones decrease with increasing bulk permeability. Another important finding is that the model calculations conducted with the effective continuum model under represent the effectiveness of condensate shedding and thereby overrepresent the magnitude of refluxing. This information will be useful in guiding the development of models of the drift-scale thermal test (see Section 5.2.3).

- A detailed description is being developed of drift-scale thermal-hydrological conditions in emplacement drifts as a function of time and location within the repository. This drift-scale thermal-hydrological description requires a three-dimensional model (or model-abstraction equivalent) that can represent both mountain-scale and drift-scale thermal-hydrological behavior. The description will be used in the total system performance assessment that supports the viability assessment (see Section 5.2.3).

- A three-step procedure has been developed to estimate permeability changes from construction-induced stress changes and from heating. Literature review shows that permeabilities are sensitive to changes in shear and normal stress, but little direct experimental data quantify the effect of stress changes or heating on permeability changes. Permeability is therefore being predicted indirectly from the effects of stress on fracture aperture and a cubic law relation between the aperture and transmissivity. Permeability of fractured rock masses is often dominated by preferential flow paths (see Section 5.2.4).

- Instrumentation for monitoring deformation and fractures in the rock was installed in the single-heater, drift-scale, and large-block tests. Instrumentation was also installed in the large block test to monitor temperatures and other important test parameters (see Section 5.2.5).

- The preliminary results of the coupled thermal-hydrological-geomechanical-geochemical responses of the heated rockmass in the single-heater test indicated that the heat moved the moisture around the heater hole. As of January 30, 1997, a small dryout region may have been created around the heater. The primary purpose of the single-heater test is to test thermal-mechanical responses of the rock mass. Therefore, to avoid interference with the thermal-mechanical holes, the boreholes for the coupled thermal-hydrological-geomechanical-geochemical processes were not located near the heater hole. Thus, the small dryout region is not well monitored, and its existence will need to be verified later, assuming it expands (see Section 5.2.5).

Results also showed that the water relocated by the heat is more diluted than the local ground water and may have only reached chemical equilibrium with the secondary minerals on the fracture surfaces. As a result, the chemistry of this relocated water is likely to be substantially different from the chemistry of J-13 water. The thermal-mechanical measurement results are not conclusive enough for assessing thermal-mechanical-hydrological couplings. A complete analysis of the data will be conducted when the heating phase of the test is completed (see Section 5.2.5).

- Experiments continued to provide data for a near-term engineering assessment of the microstructural, mineralogical, and mechanical changes in concrete and changes in associated water chemistry as a result of a repository hydrothermal cycle (see Section 5.2.6).

- Studies on microbial growth and survival are being conducted in conjunction with the large-block test, and additional studies are planned for the drift-scale test (see Section 5.2.6).

5.1 WASTE PACKAGE DESIGN (SCP S' CTION 8.3.4.2)

5.1.1 Design Activity 1.10.2.1 - Waste P... Design Development

The purpose of this activity is to develop waste package designs. This activity includes the development of conceptual designs, from which one or more concepts will be chosen for further development. Also included is the development of more detailed designs to support activities such as license application design.

Subactivity 1.10.2.1.1 - Disposal container design. The purpose of this subactivity is to develop design concepts for the disposal container itself, as opposed to the waste form that will be a part of the waste package.

Disposal container design is now focused on the following four basic container types, all of which are robust, multibarrier, metallic designs:

1. Disposal container for uncanistered fuel assemblies

2. Disposal container for canistered fuel

3. Disposal container for defense high-level waste glass pour canisters

4. Disposal container for defense high-level waste glass pour canisters and DOE-owned spent fuel

For handling and emplacement efficiencies and convenience, the disposal container designs, including those for defense high-level waste, are intended to have similar dimensions

Project emphasis has been shifted away from canistered commercial fuel because of cancellation of the multi-purpose canister effort. Accordingly, there was little analysis and design development of disposal containers for canistered commercial spent fuel during this period. Future development of a waste package for canistered Navy spent fuel is likely because the current planning assumption is that Navy spent fuel will be emplaced in the repository in canisters.

Design has shifted to analysis of the key design features discussed in Progress Report #15. Thermal, neutronics, shielding, structural, and design basis event analysis have been performed. A description of the key design features resulting from that work follows.

Disposal Container for Uncanistered Pressurized Water Reactor Fuel

In the design concept currently under consideration, uncanistered fuel assemblies would be individually placed in the disposal container at the repository, and the two disposal container closure lids welded into place. The containment barriers would consist of an outer layer of carbon steel and an inner layer of ASTM B 443 (nickel-base Alloy 625). The fuel basket would consist of plates of stainless steel-boron alloys and tubes of carbon steel. The stainless steel-boron alloy plates used in the basket construction would provide criticality control, heat conduction, and structural support. The plates would interlock to form a grid into which the square carbon steel tubes would be inserted. The basket would be supported by basket guides of carbon steel that would be welded to the inner containment barrier. The square openings in the basket would be lined with carbon steel tubes, which would provide additional structural support and an additional path for conducting heat from the fuel to the surface of the waste container. The use of aluminum thermal shunts may also be incorporated to conduct heat from the fuel to the surface of the package.

Disposal Container for Uncanistered Boiling Water Reactor Fuel

Materials and construction for the disposal container for boiling water reactor fuel would generally be similar to those of the disposal container for pressurized water reactor fuel. The grid openings in the basket would be smaller because of the smaller cross section of a boiling water reactor assembly, and because of their lower heat output, boiling water reactor assemblies do not require basket tubes or aluminum thermal shunts.

Disposal Container for Defense High-Level Waste Glass Plus DOE-Owned Spent Fuel

This disposal container is commonly referred to as the "defense high-level waste glass disposal container," but a new design is being considered to accommodate both defense high-level waste and DOE-owned spent fuel. During advanced conceptual design, the disposal container for defense high-level waste glass was designed to hold four canisters. An additional design under consideration would allow each disposal container to hold a fifth canister. In the center of the disposal container would be a circular tube. The tube could either accept a canister containing DOE-owned spent fuel or be fitted with a basket that would hold DOE-owned spent fuel. The five sealed pour canisters of high-level radioactive waste would encircle the tube. Like the uncanistered fuel disposal containers, there would be a double containment barrier, again with an outer layer of carbon steel and an inner layer of ASTM B 443 (nickel-base Alloy 625).

Canistered Fuel Disposal Containers

Because of the reduced Project emphasis on canistered fuel little analysis and design development of disposal containers for canistered commercial spent fuel occurred during this period.

Subactivity 1.10.2.1.2 - Design basis fuel. The purpose of this subactivity is to determine the design basis spent fuel for use in waste package design.

A configuration analysis (CRWMS M&O, 1997q) was prepared to assess the capacity of the waste package designs and the number of different types of package design types that would be required to handle 100 percent of the expected commercial spent nuclear fuel waste stream, should the law be changed to allow or require emplacement of all such waste at Yucca Mountain. The objective of the evaluation was to (1) determine the number of different types of waste packages needed, (2) determine the capacity of each waste package type, (3) determine the spent nuclear fuel parameters that provide the limits for each waste package type, and (4) provide reasonable confidence that the selected system of waste package types will support disposing 100 percent of the expected commercial spent nuclear fuel waste stream to be shipped to the MGDS repository. This information will help determine the scope of the waste package design efforts and will provide goals for determining the design basis spent nuclear fuel type for thermal, structural, and neutronics-criticality analysis.

Because of the large variability in spent nuclear fuel characteristics, several waste package designs will be required to accommodate all the spent fuel earmarked for disposal in the first repository. Arguably, a potential engineering solution probably exists for any spent fuel decay heat or criticality problem such that one design could accommodate all the assembly types. However, economics dictate that multiple waste package designs be tailored to portions of the waste stream (i.e., it is not cost effective to allow the most stressing 10 percent of the waste stream to drive the design for the other 90 percent). Therefore, a family of waste package designs is required, and each individual design must have a specifically designated design basis fuel. The purpose of this analysis was to develop rational waste package design and design basis fuel combinations supported by waste stream coverages, past waste package analyses, and engineering judgment

The paragraphs that follow describe the specific areas examined in the configuration analysis.

Thermal Options for Design Basis Fuel

Three total waste package heat loads (14.2, 18, and 19 Kw) were considered to determine performance and cost trends as a function of waste package thermal loading. In turn, this information will better define the appropriate design basis fuel for thermal considerations.

The thermal load on the waste package (and consequently its temperature) is most directly determined by the rate of heat generation. Both rate of heat generation and waste package temperature change with time. However, heat at time of emplacement is the single strongest determining parameter for peak waste package temperature. This important design parameter is constrained by the need to avoid cladding creep and mineral phase transformations at the emplacement drift wall. Heat at emplacement is primarily a function of age at emplacement and burnup and is a criterion for distinguishing between assembly thermal categories.

Although the waste package design basis fuel is specified on a per-assembly basis, the total waste package heat load will impact the emplacement drift structures and the surrounding rock. Previous preliminary analyses [MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b)] have indicated that initial individual waste package heat loads of around 18 kW can be tolerated assuming a reference repository thermal loading range of 80 to 100 MTU/acre. A higher initial heat, such as 19 kW, could possibly be tolerated. The study considered other system interface issues such as the 14.2 kW heat at emplacement limit imposed in the past for the conceptual multi-purpose canister design. Using a 19 kW waste package total heat load would significantly increase the risk of not meeting the repository thermal performance criteria for rock media temperatures if thermal loads in the 90 to 100 MTU/acre range are selected. A significant cost advantage is not believed to exist for this higher waste package heat load to be selected and the additional design risk accepted. It is not believed that such a cost advantage [illegible]ts. Even if the repository thermal load is kept below 85 MTU/acre as discussed in Section 4.[illegible] of this progress report, 19 Kw waste package heat load could still represent a performance problem.

<u>Criticality Options for Design Basis Fuel</u>

Analysis was performed to identify the criticality limits to be placed on design basis fuel. The criticality performance parameter, k_∞[1], and the waste package loading scenarios developed for this analysis are based on advanced conceptual design [MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b)] analysis results. All the waste package designs considered assume that principal isotope burnup credit will be accepted by the NRC. The analyses assumed that each waste package would be designed with 5-mm-thick carbon steel tubes around the fuel assemblies. When included, the neutron absorber plates would be 7-mm-thick borated stainless steel and the absorber control rods would be zirconium-clad B_4C rods. The analyses concluded that the criticality potential would not require derating a waste package or using a smaller package for pressurized water reactor spent nuclear fuel. This conclusion is based on the availability of control rods shipped with the spent nuclear fuel to reduce reactivity of relatively reactive assemblies. Because control rods cannot be inserted into boiling water reactor

[1]k_∞ is a measure of the criticality potential in a configuration with no neutron leakage. Such a configuration with a k_∞ of 1 would be critical. Because all real configurations experience neutron leakage to the surrounding environment, a real configuration with a k_∞ of 1 would be subcritical. Therefore, use of k_∞ as a performance parameter is conservative. K_∞ is the only measure of assembly criticality independent of the waste package. The spent fuel assembly analysis is performed on an assembly-by-assembly basis. However, K_∞ is an analysis tool and cannot be used to support compliance with 10 CFR 60 criticality control requirements.

assemblies, these assemblies may need to be placed in smaller capacity waste packages to achieve the required criticality control.

The analysis also concluded that the spent nuclear fuel can be segregated into three categories: that requiring no specific neutron absorbers in the waste package basket, that requiring neutron absorber plates, and that requiring the insertion of control rods (or some other special treatment) before disposal. The quantities in each of these categories were determined from the Energy Information Administration data base using the code described in Section 5.1.3, Subactivity 1.10.2.3.3. A system of waste package designs will be developed using combinations of these waste types, each with a given capacity and criticality control rating.

The above analysis constructed only undegraded fuel. A future analysis (as yet not scheduled) will examine degraded configurations.

Package Loading Options for Design Basis Fuel

A secondary purpose of the design analysis was to determine the cost effectiveness of derating the large waste package (i.e., less-than-fully loading containers if high thermal-output fuel is to be loaded) versus using a second, smaller-capacity waste package for the assemblies that could not be placed in the large-capacity disposal container. Also investigated was the impact of the total capacity of the large waste package on system cost. To bound these possible options, the following loading scenarios were considered: large packages with derated secondary designs; large package designs, along with smaller waste packages designed specifically for smaller capacities; and all smaller capacity waste package designs. The results of these evaluations are discussed in the paragraphs that follow.

Recommended Design Basis Waste Package Configurations

Each combination of the above waste package design options (thermal, criticality, and loading) was considered and compared on the basis of design feasibility and total waste package production cost. Several different waste streams (both typical and bounding) were considered. The recommended design basis waste package system configuration is presented in Table 5-1, and the rationale supporting this selection is provided in reference design analysis (CRWMS M&O, 1997q).

Each waste package type in the table is characterized by a design basis heat and criticality potential range. Coverage ranges in Table 5-1 indicate the resulting number of waste packages of that type and what percentage of the assemblies (pressurized or boiling water reactor) that are captured by that waste package type. A range is reported because the coverage varies with the waste stream assumed. Different thermal and criticality options are included in the description of each waste package type.

For example, a waste package that would hold 21 pressurized water reactor assemblies with neutron absorber plates would be able to load pressurized water reactor fuel assemblies with a thermal output of up to 850 W and a k_∞ of up to 1.13. To load the entire waste stream of

Table 5-1. Design Basis Waste Package System Configuration Waste Stream Coverage

Waste Package Types	Design Basis Heat Range (W)		Design Basis Criticality Range		Coverage Range	
	Hmin	Hmax	k_s min	k_s max	Number of Waste Packages	Percent of Assemblies that could be Emplaced in each Waste Package Type
21 pressurized water reactor - no absorber (base thermal & criticality case)	0	850	0.00	1.00	1375 to 1835	26.9 to 40.6
21 pressurized water reactor - absorber plates (criticality option 1)	0	850	1.00	1.13	2399 to 3596	53.1 to 58.1
21 pressurized water reactor - absorber rods (no absorber plates) (criticality option 2)	0	850	1.13	1.45	119 to 257	2.6 to 4.1
12 pressurized water reactor - no absorber (thermal option 1)	850	1370	0.00	1.02	80 to 850	1.0 to 7.7
12 pressurized water reactor - absorber plates (long waste package to accommodate South Texas Project fuel)	0	1370	0.00	1.13	150 to 272	1.9 to 2.5
44 boiling water reactor - no absorber (base thermal & criticality case)	0	400	0.00	1.00	695 to 997	24.6 to 30.3
44 boiling water rector - absorber plates (criticality option 1)	0	400	1.00	1.37	1942 to 2704	68.2 to 74.6
24 boiling water reactor - thick absorber plates (thermal option 1 and criticality option 2)	0	520	0.00	1.54	40 to 197	0.8 to 2.8

Note: *The quantities in each of these categories are determined from the Energy Information Administration data base using the code described in Section 5.1.3, Subactivity 1.10.2.3.3 of this progress report* assemblies that would fall within this category would require 2399 to 3596 such waste packages. This type of waste package could emplace 53.1 to 58.1 percent of the waste stream.

This table shows the recommended suite of waste package designs that is expected to allow emplacement of all commercial spent fuel that falls within design basis limits. The small amount of fuel that falls outside the design basis limits will be dealt with on a case-by-case basis. This approach is expected to be an optimal solution in terms of cost and flexibility.

Subactivity 1.10.2.1.3 - Waste Package Emplacement Support. The purpose of this activity is to develop a preliminary design for the in-drift emplacement supports for the waste package. The preliminary designs have been developed using thermal and structural analyses and interface requirements with preliminary emplacement drift invert and lining designs. The emplacement support design consists of two main components: a pier and a support. A sketch of the design is provided in Figure 5-5.

The waste package pier would interface with the precast concrete drift invert. The pier would be constructed primarily of concrete because it would be primarily loaded in compression. The concrete would be surrounded by a thin carbon steel shell to provide better interfaces with the invert and to slow the drying rate of the concrete that may be increased by the relatively high temperatures in the drift. The steel plate on top of the pier would be thicker than the other steel plates on the pier to help distribute loads in the concrete.

The waste package support would sit on top of the pier. Pins on the bottom of the support would aid in placing the support on the pier and in holding the support in place. Pipes welded to the pins would act as support columns, and rectangular tubing welded in a vee shape would make up the support saddle to hold the waste package. The waste package support would be constructed entirely of carbon steel.

Forecast: The disposal container designs will be analyzed in detail in support of the 1998 viability assessment. Few significant changes to the design are expected before the viability assessment. The analysis of internal criticality and the effects of basket degradation will be revised. Design concepts for an additional barrier or "drip shield" will be developed and evaluated during fiscal year (FY) 1997.

5.1.2 Design Activity 1.10.2.2 - Design Tools

The purpose of this design activity is to develop, verify, and validate the computer design tools for waste package design.

Plans were revised for verifying and validating new versions of the thermal and structural analysis tool ANSYS. ANSYS version 5.2 was to have been verified and validated during the reporting period, but it was determined to be more cost effective to continue to use version 5.1, and then upgrade directly to version 5.4.

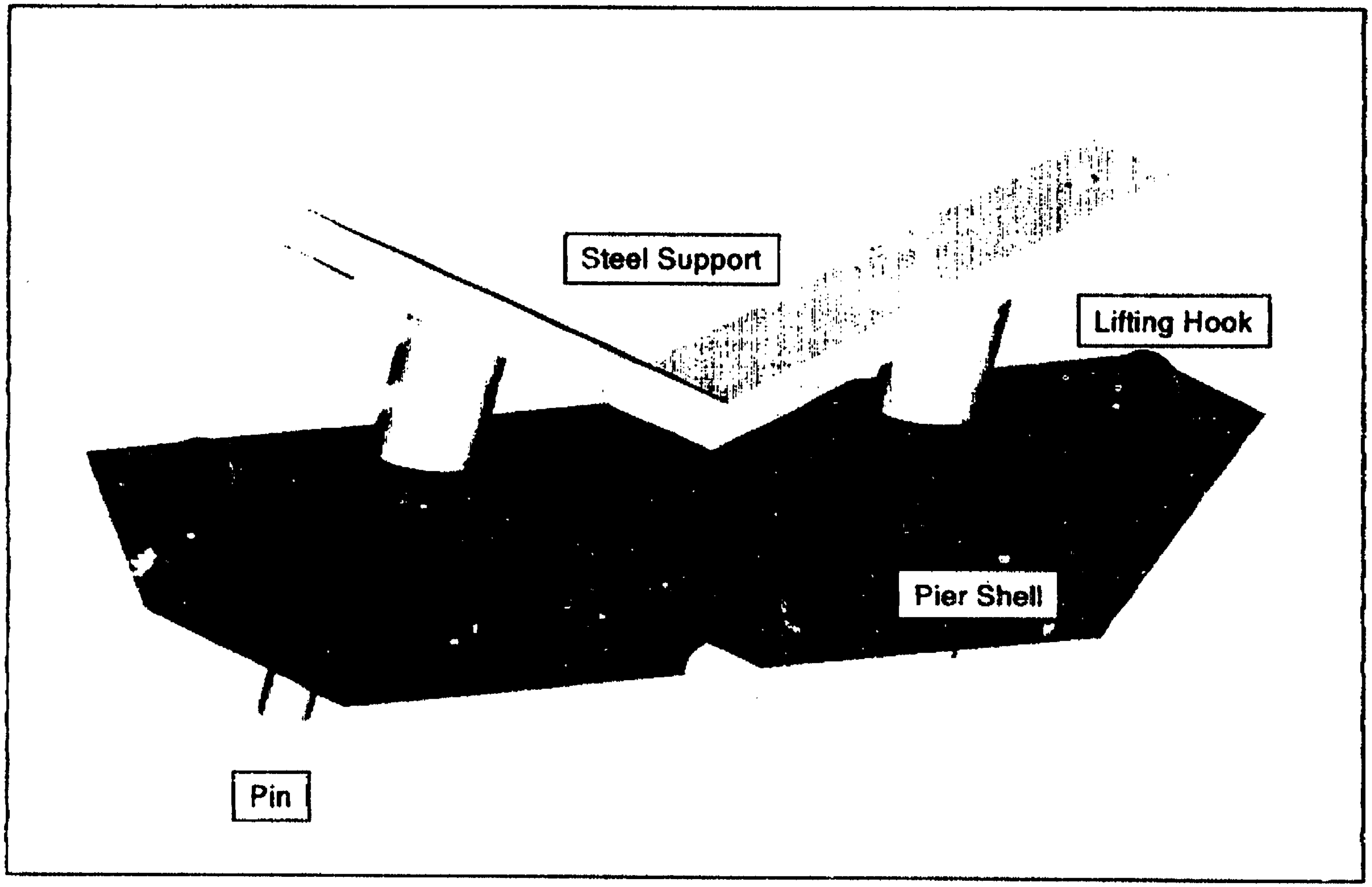

Figure 5-5. Waste Package Support Concept

Two changes affecting the neutronic computer code for waste package design occurred. An addendum to the MCNP 4A software qualification report was written to add ENDF/B-VI cross sections to the currently approved MCNP library. The use of the MCNP 4A with the ENDF/B-VI cross sections is now approved for use in quality-affecting activities. The SCALE (Standardized Computer Analyses for Licensing Evaluation) version 4.3 code package was verified and validated in accordance with the appropriate procedures. SCALE 4.3, which is now approved for use in quality-affecting activities, will replace the SCALE 4.2 package, which will be retired.

Forecast: During the second half of FY 1997, the DORT computer code and BUGLE-93 cross section library will be verified and validated for quality-affecting design work. The MCNP 4.2 and SCALE 4.2 code systems will be retired during the next reporting period. The thermal and structural analysis tool ANSYS will be upgraded to version 5.4. These computer design tools wi' support future design products such as those described in Sections 5.1.1 and 5.1.3 of this progress report.

5.1.3 Design Activity 1.10.2.3 - Design Evaluations

The purpose of this activity is to produce evaluations of significant issues pertinent to successful waste package design.

All the thermal studies discussed in this section neglected convection in the drift. Convection is believed to be insignificant to peak cladding temperatures. Also, percolation flux is not expected to impact peak cladding temperature and therefore was not a variable or assumption of interest in these studies.

Subactivity 1.10.2.3.1 - Thermal. The purpose of this subactivity is to perform analyses of thermal performance parameters that affect waste package design.

Thermal design efforts have advanced in three main areas: (1) evaluating the repository and emplacement drift thermal behavior and its impact upon waste packages, (2) evaluating waste package thermal conditions with regard to meeting the licensing requirements, and (3) evaluating and designing the waste package support and invert based on its thermal and structural performance under nominal MGDS repository conditions.

Emplacement-Scale Thermal Analyses

The first step in the thermal evaluation of the waste package is to determine the time-dependent response of the repository to the decay heat of the emplaced waste packages. This emplacement-scale evaluation must consider that the waste package both affects (through thermal loading) and is affected by the conditions of its near-field environment. Although a given thermal (mass) loading is typically characterized by a single number, such as 83 MTU/acre, the thermal response of the repository depends on the heat generation as a function of time in the various waste packages, which have different values of this important process variable. The heat generation in turn depends on the characteristics of the waste stream such as spent nuclear fuel age, receipt rates, delivery scenarios (youngest-fuel-first versus oldest-fuel-first), waste package size, emplacement spacing, and design basis fuel.

As reported in the previous progress report, emplacement-scale evaluations have been performed to support systems studies considering both emplacement drift backfill and thermal loading and to advance the waste package design effort. The results of these evaluations indicate waste package and drift wall temperatures depend in large part on the assumptions used to estimate the average waste stream heat loads and the variability in heat loads within the waste stream. These evaluations indicated a need to investigate the impact of waste stream variability on waste package capacities and design basis heat output. The subsequent evaluation of the impacts of waste stream variability on waste package design basis fuel determination, including individual waste package heat outputs, is described in Section 5.1.1, Subactivity 1.10.2.1.2.

During the current reporting period, further emplacement-scale evaluations were performed to determine the impact of higher thermal loadings (100 MTU/acre) and selective waste package placement. The near field with multiple waste packages was evaluated (CRWMS M&O, 1997r) to investigate the thermal effects of reduced waste package spacings and the impact of individual waste package heat load variability. Assuming variable spacings from the MGDS Advanced Conceptual Design Report (CRWMS M&O, 1996b) to account for the waste loading of each waste package, a multiple waste package evaluation at 100 MTU/acre was performed and compared with previous evaluations at 83 MTU/acre. A thermal (mass) loading of 100 MTU/acre is considered to be an upper bound (CRWMS M&O, 1996c) for the potential repository, and was evaluated to ensure design flexibility and compatibility at the limiting thermal loading.

At the 100 MTU/acre loading, the estimate of peak cladding temperatures within the waste package with 21 design basis pressurized water reactor assemblies increased by only 4°C as compared to a loading of 83 MTU/acre, but the estimated peak drift wall temperature increased roughly 20°C. Calculated peak drift wall temperatures did, however, remain below the limit of 200°C assuming nominal advanced conceptual design spacings. The highest drift wall temperature predicted at 100 MTU/acre was 188°C at 40 years after emplacement. This temperature level is close to the assumed drift wall temperature limit. An upper limit of 85 MTU/acre described in Section 4.1.6 of this progress report is based on limiting peak temperature at the average top of the underlying zeolitic layer.

More detailed evaluations of peak internal (and cladding) temperatures, and the effect on the waste package design, are discussed below in the section on waste package scale thermal analyses.

Evaluations, reported previously, indicated that waste package spacings required to achieve a "line loading" would result in temperatures signi[illegible] above the assumed design goals (limits). An important caveat for the previous evaluati[illegible] is that the arrangement of waste packages was not specified in a way to minimize peak temperatures. For example, two waste packages containing 21 design basis pressurized water reactor spent fuel assemblies were always assumed to be adjacent to each other. Therefore, the results are considered conservative. A follow-up evaluation was performed to determine the impact of this caveat on the conclusions of the previous work. Assuming a waste package spacing of 0.1 m (a scenario that was previously shown to be more than 50°C above thermal limits) the model was rearranged to always separate waste packages with design basis fuel with cooler high-level waste packages in between. As expected, the average calculated drift wall temperatures along the drift remained unchanged. However, by separating high level waste packages with the hotter design basis fuel, peak temperatures near the hottest package were lowered by nearly 10°C. Estimates of drift wall temperatures, both peak and average, remained significantly above the thermal goal. High peak drift wall temperature, therefore, remains the primary impediment to the feasibility of the line loading concept.

Waste Package Scale Thermal Analysis

Specific finite-element models of waste packages for uncanistered fuel have been developed to address potential changes in the waste package designs. The representative waste package designs being evaluated include capacities of 21 pressurized water reactor fuel assemblies and 44 boiling water reactor fuel assemblies. The thermal analysis at the waste package scale focuses on the internal structures of the waste package, which includes interlocking criticality control plates, fuel assembly tubes, basket support guides, and the potential thermal shunts.

The thermal evaluation of the waste package with 21 pressurized water reactor uncanistered fuel assemblies considered the following variables in the design: (1) thermal conductivity of the assembly tubes, (2) tube thickness, and (3) use of aluminum thermal shunts.

The thermal conductivity of assembly tubes was analyzed to compare the effect of tube materials on the waste package performance. Carbon steel A 516, AISI-SAE 1008, and aluminum alloy Al 6061 were chosen for analysis. The results showed that changing material from carbon steel A 516 to a higher conductivity carbon steel like AISI-SAE 1008 would improve the waste package thermal performance. However, the peak temperature reduction would be less than 15°C. For aluminum tubes, the peak fuel temperatures would decrease more than 60°C. However, the mechanical strength and the cost of the material must be considered as well as the heat transfer properties.

Different thicknesses (5 mm, 6 mm, and 7 mm) of low-cost A 516 tube material were analyzed. The results indicated that the peak fuel temperature would exceed its limit unless a thickness of 7 mm or greater is used. This thickness would significantly increase the waste package diameter and therefore the cost of the waste package.

Another option is to use an alternative construction of the interlocking plates to include four additional aluminum alloy thermal shunts in the basket. This construction takes advantage of high thermal conductivity of aluminum to efficiently remove the heat from the center of the waste package. The results showed that this design can satisfy the cladding temperature limit of 350°C and that it also maintains a reasonable waste package size.

Waste Package Support/Invert Thermal Analysis

A thermal calculation was performed for the waste package support-invert design to provide thermal parameter information for the preliminary waste package support design under nominal MGDS repository conditions. Temperatures of the waste package surface and the support components were determined by using finite-element analysis. The analysis began by modeling a full three-dimensional waste package-support repository model with simplified support structure. Detailed design of the waste package support was modeled to determine the effective conductivity of the support structure. The results showed that the maximum waste package surface, support, and pier temperatures would be around 200°C after 10 years emplacement. The temperatures at 1 m into the rock would be below 200°C. The temperature along the concrete liner would range from 174° to 157°C at the time of peak temperatures. This result implies that the waste package would be likely to meet the temperature limitation if the proposed design of waste package support is applied in the repository. The results also suggested that the design of the concrete liner and the invert should consider the temperature effect on the material and structural strength.

Subactivity 1.10.2.3.2-Structural. Two major activities were included in this reporting period. One activity focused on the preliminary design of the waste package support and pier. To support this activity, a design analysis was completed and documented in Waste Package Support and Pier Static and Seismic Analyses (CRWMS M&O, 1997n). The objective of this analysis was to determine appropriate dimensions and materials for the waste package support and pier using structural requirements. The waste package support and pier resistance to the weight of the waste package under static and seismic load conditions were evaluated. The results of this report will provide input for the waste package support and pier drawings. The document was in design review at the end of the reporting period.

A letter report was also being written for the design activity based on the structural and thermal analyses of the support and pier assembly. The objectives of the letter report are to present the preliminary designs for the waste package support and pier, summarize results of analyses performed on the designs, assess the pier-pier liner interface, and describe an alternative design for an all steel pier.

The second activity focused on the waste package structural analyses. Development of the document Drop Analyses of Uncanistered Fuel Waste Package Designs (CRWMS M&O, in prep.[e]) was in progress at the end of the reporting period. The purpose of these analyses is to help determine waste package component dimensions. The component dimensions are required to show the adequacy of the uncanistered fuel waste package design with stainless steel-boron neutron absorber plates under loading encountered during waste package drop events. The drop events evaluated are 2-m drops onto an essentially unyielding surface. The objective of this analysis is to determine the proper dimensions for waste package components. The waste package dimensions will become design input for waste package technical drawings.

A supporting design analysis was also in progress this reporting period and is documented in the analysis entitled "Uncanistered Fuel Waste Package Static Loads, Thermal Expansion Loads, and Internal Pressure Analysis" (CRWMS M&O, in prep.[f]). The objective of this analysis is to determine appropriate dimensions and materials for the uncanistered fuel waste package designs from structural requirements. This document will contain the waste package resistance to the weight of the waste package under static, thermal expansion, and internal pressure loads, and will provide input for waste package technical drawings.

ANSYS version (V) 5.1, a finite-element analysis computer code, was used in the structural analyses. The finite-element solution is based on the forces and moments developed within the solid model due to external forces and moments, these are included in all finite-element solutions. The stresses obtained from the finite-element analysis were compared to the material yield strengths for the tube, pipe, and the plate; compressive ultimate strength of the pier is compared to the maximum bearing stress. Since the subject components are not parts of a pressure vessel, ASME code sections are not directly applicable to the design analysis of the waste package support and pier. Instead, a common engineering approach was taken by using the distortion-energy-theory, which is confirmed to be "the best theory to use for ductile materials." Details of this theory were provided within the report.

Waste Package Support and Pier Preliminary Design

Development of a support structure for in-drift emplacement of the waste packages is in progress; Figure 5-5 shows the preliminary design. The waste package support structure design is intended for use with repository designs that use concrete lining of the emplacement drifts. A modular design was developed for the waste package support assembly. The assembly consists of a waste package pier and a waste package support. A modular design is desirable because it allows flexibility in waste package placement in the drifts and individual component replacement if the support structure is damaged in a waste package handling accident. In such an accident, avoiding support structure damage is not as critical as preventing a breach of the waste package. The support structure is therefore designed to yield if a waste package handling accident occurs.

The pier and support would hold the waste package off of the invert to allow for a sorptive or filter bed below the waste packages if needed. The bed may be used to trap radionuclides after the waste package has degraded. Holding the waste package above the invert would also aid in preventing water from contacting the waste package if water in the drift would begin collecting on the invert.

Two designs have been considered for use with the repository designs that use a precast concrete invert. These two designs were determined by the lower and upper ranges for gantry wheel size (300 and 600 mm in diameter, respectively). The size of the gantry wheel affects the height of the haunches on the precast invert and waste package pier. The only difference between the two designs is the height of the waste package pier. Because of the modular design, the same waste package support could be used in both designs.

The waste package support would be fabricated from two pieces of rectangular steel tubing, two pipes, and two round bars. The two pieces of rectangular steel tubing would be welded together to form a vee shape. Then the two pipes would be welded to the tubes. Round bars would be engaged into the pipes and welded from inside the pipes. These bars would serve as pins to position the supports on the pier and prevent tipping of the support. The pipes would serve as columns to transmit load from the tubes to the top plate of the pier. For this analysis, a standard size of rectangular tubing was selected to reduce manufacturing costs. The steel types chosen were ASTM A500 Grade B for the rectangular tube, ASTM A501 for the pipe, and ASTM A36 for the round bar. These carbon steels are standard materials for these shapes. The sizing of the support depended on the required strength, the available space in the drift, and the desired spacing between the invert and the bottom of the waste package.

The waste package pier is fabricated from steel plates, steel pipe, steel bars, concrete, and rebar. Steel would be added to the pier mainly to improve interfaces with other components in the drift. The steel plates would be welded to form the waste package pier shell. The steel chosen was ASTM A36 for the plates. This carbon steel is readily available in the shapes specified. The shell would serve as the form for the concrete which is cast in the shell. The shell would prevent chipping of the concrete and would slow drying out of the concrete from the relatively high temperatures in the emplacement drift. The steel shell would also provide a better interface between the pier and the concrete invert. The rebar design has not yet been developed, but small rebar sections have been included in the design sketches as lifting hooks. Holes on the top surface of the concrete and the top plate would allow the attachment of the waste package support. The bars on the bottom of the pier would be used for positioning on the invert, and the half section of pipe on the bottom of the pier would be provided to allow for drainage of any water before it contacts the waste package.

A three-dimensional half-symmetry finite-element model of the waste package support structure has been developed to perform a static analysis on the system. The waste package weight was applied as an external load on the waste package support at the locations of contact with the waste package. The weight of the waste package support and pier system was also taken into account by the use of gravitational acceleration in the finite-element model.

The maximum-distortion-energy theory was used to determine the initiation of yield in the materials. This theory is based on a comparison of the material yield strength with the maximum equivalent stress (von Mises stress) observed in the material. The results showed that the maximum equivalent stress magnitude would occur on the inner surface of the tube side wall. The maximum equivalent stress magnitude in the tube would be less than the yield strength of the tube (see Table 5-2). Therefore, the static load does not cause yield in the tube.

Table 5-2. Waste Package Support and Pier Static Analysis Results[a]

Support and Pier Structural Components	Maximum Equivalent Stress	Tensile Yield Strength	Bearing Stress (compression)	Compressive Yield Strength	Compressive Ultimate Strength
Tube	172.5	317.0	NA[b]	700.0	NA
Pipe	89.0	290.0	NA	700.0	NA
Plate	40.3	248.0	NA	700.0	NA
Pier	NA	NA	16.87	NA	34.5

[a]All stress magnitudes are in MPa
[b]NA = not applicable

A similar comparison was made for the rest of the support structure components. For the pipe, the maximum equivalent stress magnitude was calculated to be on the outer surface of the pipe, in the region of contact with the tube. For the plate, the maximum equivalent stress magnitude was found on the top surface of the plate in the region of contact with the pipe. In either of these components, the maximum calculated equivalent stress magnitudes were less than the material yield strength (Table 5-2). Therefore, the static load would not be expected to cause permanent deformation on these support components.

Temperature-dependent material properties for ASTM A500 cold-formed steel, ASTM 501 hot-formed steel, and ASTM A36 carbon steel were not available for structural analysis. For this reason, room temperature (20°) material properties were used in these analyses. The properties of carbon steels for which temperature dependence is known change little for the temperature range of interest. Therefore, for this initial set of calculations, use of room-temperature properties is considered adequate.

The compressive strength of the carbon steel is higher than tensile strength. Thus, no permanent deformation would be expected to take place due to compression (Table 5-2).

The structural analyses of the waste package pier also revealed that the maximum bearing stress magnitude in the pier would be less than the compressive ultimate strength of the concrete (Table 5-2).

A seismic factor of 1.66 was applied to the load of the static finite-element analysis to obtain a preliminary result for the seismic design of the support and pier structure (see Table 5-3). The results were compared with the material yield strengths to determine locations of any permanent deformations in the system. The resulting maximum stresses in the support structure were less than the yield strength of the materials. Therefore, structural performance of the waste package support system components was determined acceptable under seismic loading.

Table 5-3. Waste Package Support and Pier Seismic Analysis Results[a]

Support and Pier Structural Components	Maximum Equivalent Stress	Tensile Yield Strength	Bearing Stress (compression)	Compressive Yield Strength	Compressive Ultimate Strength
Tube	286.7	317.0	NA[b]	700.0	NA
Pipe	147.8	290.0	NA	700.0	NA
Plate	66.9	248.0	NA	700.0	NA
Pier	NA	NA	28.0	NA	34.5

[a]All stress magnitudes are in MPa
[b]NA = not applicable

Structural evaluations of the waste package support and pier design presented in the structural design analysis document (CRWMS M&O, 1997n) showed that the dimensions and material properties are acceptable and can be used to develop technical drawings of the subject structural components.

If the Project should decide that concrete will not be used in the emplacement drifts, an all-steel pier would likely be designed. This design would be such that the same waste package support can be used as is used in the concrete pier design. The all-steel pier design may consist of a steel plate welded on top of a frame constructed of wide flange, I-beams. Four beams angled slightly from vertical would be used as legs to support the load. They would be angled to provide a wider, more stable base for the pier. The lower ends of the four beams would be linked by four horizontal beams to prevent spreading or closing of the space between the legs. Components would be sized appropriately to hold the weight of the waste packages.

<u>Waste Package Structural Analyses</u>

Supporting design analyses for the analysis titled Drop Analyses of Uncanistered Fuel Waste Package Designs are being performed. These analyses will determine the component dimensions required to show that the uncanistered fuel waste package design with stainless steel-boron neutron absorber plates will perform adequately under loading encountered during waste package drop events. This design analysis is currently in progress. The drop events to be evaluated are 2-meter drops onto an essentially unyielding surface. The waste package orientations during the impact will include horizontal drops with the basket members at 90 and 45 degrees from the unyielding surface. The critical basket orientation is expected to be one of these two configurations. The more critical orientation of these two will be determined for the horizontal orientation drop of the 21 pressurized water reactor assembly and will be assumed to be the same for all other waste package drop orientations that will be analyzed later. This analysis will determine the proper dimensions for the waste package components so that these dimensions may be formally passed on to drafting as input for drawings of the design for 21 pressurized water reactor uncanistered fuel design.

Waste package resistance to the weight of the waste package under static, thermal expansion, and internal pressure loads is being determined to support the Uncanistered Fuel Waste Package Static Loads, Thermal Expansion Loads, and Internal Pressure Analysis. The effects of these loads will be analyzed individually as well as collectively to obtain the most critical stress magnitudes on the waste package. This calculation will also provide input for the waste package technical drawings.

Subactivity - 1.10.2.3.3 - Criticality. The purpose of this subactivity is to perform disposal criticality control analyses and to determine the impact of criticality control issues on waste package design. The criticality activities performed during this reporting period have consisted of work in the following areas: developing inputs in support of the disposal criticality analysis methodology reports, developing the approach for integral principal isotope burnup credit, supporting the differential actinide-only burnup credit effort, evaluating designs, meetings with the NRC staff to discuss the disposal criticality analysis methodology reports, and other supporting efforts concerning neutronics issues for disposal. The work performed in each of these areas is discussed below in more detail.

Disposal Criticality Analysis Methodology Reports

The development of the disposal criticality analysis methodology reports (technical and topical) is a major, multiyear task that continued during this reporting period. The Disposal Criticality Analysis Methodology Topical Report (to be developed in 1998) is intended to present the methodology for performing disposal criticality analysis (including the use of burnup credit) for any fissile waste form, waste package design, and proposed repository. The initial issuance of the report will focus on commercial light water reactor fuel, and additional fuel types will be covered in amendments or revisions to the topical report. Before the release of the topical report, the disposal criticality analysis methodology is being developed and presented in a technical report, Revision 0 of which was issued last reporting period. During this reporting period, the NRC staff reviewed Revision 0 and provided comments and questions. An Appendix 7 Meeting was held with the NRC staff to discuss the comment and questions and the methodology. The meeting primarily focused on addressing NRC staff comments on the proposed approach to validate criticality and neutronics models. Another Appendix 7 Meeting with the NRC staff is planned for next reporting period to further discuss the technical report and the methodology it describes.

Revision 0 of the technical report identified areas in which additional supporting information and analyses are required to complete the methodology. During this reporting period, technical information was being developed to support Revision 1 of the technical report. The supporting information being developed includes evaluations of commercial reactor criticality data, critical benchmark data, and chemical assay data. Other supporting information is being developed for the configuration determination and grouping. This work is ongoing.

Burnup Credit

Seeking credit for burnup means seeking regulatory approval to account for the reduced reactivity or criticality potential of spent nuclear fuel as compared with the same fuel before irradiation. Burnup credit is a major aspect of disposal criticality control, and work to obtain it remained a major activity in this reporting period. Burnup credit is not an on or off option. Different levels or amounts of burnup credit are being sought, and the amount of burnup credit depends on the set of isotopes included. For transportation applications, burnup credit is being sought for a set of 10 isotopes called the "actinide only isotopes." For disposal activities, burnup credit is being sought for a set of 29 isotopes, called the "principal isotopes," which includes the 10 isotopes from the actinide-only set. The different isotope sets represent the different amounts needed for the two different applications. The principal isotope set accounts for more reduced reactivity than the actinide-only set.

Besides the different isotope sets, there are currently two approaches being pursued to develop burnup credit, differential and integral. The differential burnup credit approach is an ongoing activity supporting transportation applications. A topical report using actinide-only differential burnup credit remains under review by the NRC.

The integral approach is the main focus of the disposal activities. This approach uses integral commercial reactor criticality experiments as the bases for demonstrating the ability of the approach to correctly predict the reactivity of systems with spent/irradiated nuclear fuel. Commercial reactor criticality data are being used in conjunction with chemical assay data and benchmark criticality data to develop the approach. Chemical assay data are used as an acceptance criterion to demonstrate the conservatism of the isotopics model portion, while the commercial reactor criticality and benchmark criticality data are used to determine biases and uncertainties for the criticality model portion of the disposal burnup credit approach. Commercial reactor criticality evaluations continued this reporting period. This work is ongoing.

The supporting data being developed for each approach is, when appropriate, being used for the other approach. These activities are ongoing.

Additional Criticality Evaluations

Criticality evaluations continued for the 12 and 21 pressurized water reactor uncanistered fuel, 44 boiling water reactor uncanistered fuel and the 5 defense high-level waste glass waste package designs. Material modifications and small dimensional changes made for noncriticality reasons are being considered. These activities are ongoing, and will continue into future reporting periods.

Methodology

Previous evaluations of the possibility of degraded waste package criticality have used computer codes to track the concentrations of fissile and neutron absorber species and estimate k_{eff} for the most likely geometric configurations. Specifically, two general codes were developed: one for commercial spent nuclear fuel in a waste package with partly or completely degraded basket, and the other for degraded immobilized plutonium waste forms (glass or ceramic) with fissile material collected in clay precipitate at the bottom of the waste package. These codes are now being combined and extended to cover the four general categories of waste forms expected to be received at the repository that have significant criticality potential: (1) commercial spent nuclear fuel, (2) DOE-owned spent nuclear fuel, (3) immobilized plutonium waste forms, and (4) mixed-oxide spent nuclear fuel using plutonium from decommissioned weapons. This program will also track all the successive stages of waste form degradation and the resulting possible criticality locations: internal, near-field external, and far-field external. The design document for the new combined code system has been written and is being reviewed to ensure the accuracy of the abstractions of the geochemical and fluid mechanical processes to be captured in this program.

Consequences of a Hypothetical Criticality

Previous evaluations of consequences of a criticality in a repository estimated the increase in radionuclide inventory assuming a steady state "reactor" operating at precisely $k_{eff}=1$ for thousands of years. Actually, a critical configuration would experience a k_{eff} increase slightly beyond 1 for a brief transient period until negative feedback mechanisms would take over (typically by evaporating moderator or solution containing fissile material) and drop the k_{eff} below 1. This sequence of overshoot and fallback would be repeated if the fissile material or moderator continue to flow into the critical configuration. Furthermore, there is a possibility (for certain very unlikely configurations) that the initial feedback when k_{eff} increases beyond 1 would be positive, and the negative feedback would not take effect until the k_{eff} had overshot the critical value of 1 by up to 20 percent. The Project has obtained or is in the process of obtaining the following three codes that allow estimating the increased radionuclide inventory and the energy release during such transient periods:

1. RELAP5 (REactor Leak And Power excursion). This is the standard code for reactor transient analysis, with coupled neutronic, hydraulic and thermal capability, and the capability to model detailed assembly geometry. The code, developed under the auspices of, and certified by, the NRC, is directly applicable to the internal criticality configuration for intact spent nuclear fuel. The code has been acquired from the vendor.

2. NARK (Nuclear Reactor Dynamics Model). This code was developed at Sandia National Laboratories for use on the Waste Isolation Pilot Plant and for DOE spent nuclear fuel criticality. The code has the capability to model much more general configurations than RELAP5, but with less detail. NARK will be applicable to external criticality or severely degraded internal criticality. The Project is in the process of acquiring the code.

3. MRF3. This code was developed at Los Alamos National Laboratory to model potentially autocatalytic configurations, and to estimate the partitioning between thermal and kinetic energy release. This code is similar in capability to NARK but can

also model the response of a confining medium. MRKJ will be applicable to external criticality of highly enriched fuel. The Project is discussing how to acquire the code with the staff at Los Alamos National Laboratory.

Other Activities

An updated version of the SCALE neutronics computer code packages (SCALE4.3) was validated and verified according to the Quality Administrative Procedures during this reporting period. The code is being used to support the neutronics evaluations performed for the technical and topical reports and for the designs.

Subactivity 1.10.2.3.4 - Radiation effects. The purpose of this subactivity is to examine radiation effects on waste package degradation. Shielding evaluations continued that examined the details of the radiolysis effects inside the waste package. The radiation and radiolysis effects from alpha radiation from commercial fuel pellets in failed fuel (fuel with ruptured cladding) were evaluated for pressurized water and boiling water reactor fuel. Evaluations of beta and gamma radiation effects on radiolysis had previously been evaluated. The evaluations feed models of internal waste package material degradation.

Subactivity 1.10.2.3.5 - Identification of waste package preclosure design basis events. The purpose of this activity is to review the MGDS design to identify a bounding list of credible preclosure design basis events for the waste package. (Design basis events are, by definition in 10 CFR 60, applicable only to preclosure.) This is a new activity, reported for the first time this reporting period, that is related to Activity 2.7.1.1, reported in Section 4.2.1 of this progress report. The frequency and severity of events involving the waste package, were initially assessed in early 1996 based on the advanced conceptual design (CRWMS M&O, 1996b) and reported in the Waste Package Off-Normal and Accident Scenario Report (CRWMS M&O, 1996u). Additional analysis has been performed in early 1997 using the updated assumptions and MGDS design for viability assessment. Complete details of this analysis are provided in the Waste Package Design Basis Events QAP 3-9 design analysis (CRWMS M&O, 1997s). The method used for this analysis involves the following four steps:

1. The Preliminary MGDS Hazards Analysis (CRWMS M&O, 1996q) and the Waste Package Off-Normal and Accident Scenario Report (CRWMS M&O, 1996u) are reviewed to identify internal and external events that have the potential for adversely affecting the performance of the waste package. Other sources of information on the current repository surface and subsurface design may also be used in the identification, screening, and characterization of internal events. In addition, the NRC Standard Review Plan for Dry Cask Storage Systems (NRC, 1997b) provides guidance on the types of events that are expected to be evaluated in a license application.

2. Preclosure events are screened for applicability to waste package design. An event may be screened from further consideration if it meets one of the following criteria:

   - The event was screened in the Preliminary MGDS Hazards Analysis (CRWMS M&O, 1996q). The Preliminary MGDS Hazards Analysis eliminated some external events from further consideration because they were either not applicable to the Yucca Mountain site or not applicable to the preclosure phase of the MGDS.

   - The event cannot directly affect the performance of the waste package. This may be because either (a) the waste package is contained within another system, structure, or component that has been assigned the function of protecting its contents from such an event; or (b) the event results in the disruption of a service that is not required by the waste package to continue to perform its functions.

   - The event has an estimated frequency of occurrence of less than 1×10^{-6} events per year. The Waste Package Off-Normal and Accident Scenario Report (CRWMS M&O, 1996u) initially estimated the frequency of events such as a spent fuel assembly drop, waste package drop, waste package slap down, transporter derailment and runaway, fire, flooding, rockfall, and missile hazards. The current analysis updates these frequencies with new information, as well as estimates frequencies for additional events such as misloads exceeding the thermal or criticality design basis of the waste package, and the occurrence of through-wall manufacturing defects. Events with frequencies less than 1×10^{-6} events per year are not considered credible and are screened from further consideration as preclosure design basis events, as per the section-by-section analysis of 10 CFR 60.136 (61 FR 64257).

3. The severity of events (from a waste package perspective) which are not eliminated from further consideration under item 2 is estimated. In addition, the general type of analysis that will be required (structural, thermal, or criticality) to determine the effect of the event on waste package performance is identified.

4. Similar events from item 3 are grouped for the purpose of identifying a bounding event for each group based on severity.

Table 5-4 provides the bounding list of credible preclosure design basis events for the waste package identified in the Waste Package Design Basis Events analysis (CRWMS M&O, 1997s).

Source Terms Analysis for Design Basis Events

This activity was performed in mid-1996 to define the radiological source terms to be used in the consequence analyses of design basis events involving pressurized or boiling water reactor spent nuclear fuel, or defense high level waste glass canisters. Because this work was not reported in previous progress reports, it is reported here. The results of this activity represent

Table 5-4. Bounding Waste Package Design Basis Events

Analysis Type	Event Group	Magnitude and Severity
Structural	Falling Objects - Side Impact	10 metric tons rock falling 3.1 m
	Falling Objects - End Impact	2.3 metric tons falling 2 m
	Vertical Drops and End Collisions	2 m drop
	Horizontal Drops and Side Collisions	2.4 m drop
	Puncture Hazards	1.9 m horizontal drop onto support or 2.4 m horizontal drop onto pier, whichever is worse.
	Tip-over and Slap-down	Waste package tips over from a vertical position and slaps down onto a flat surface.
	Seismic Activity	Maintain structural integrity and prevent tip-over for 0.66 peak and ground acceleration
	Missile Hazards	0.5 kg missile at 5.7 m/s
	Fuel Rod Rupture/Internal Pressurization	See CRWMS M&O (1997s) for internal pressure as a function of gas temperature
Thermal and Structural	Thermal Stresses and Peak Waste Form Temperature	Exposure of whole waste package for not less than 30 minutes to a heat flux not less than that of a radiation environment of 800°C with an emissivity coefficient of at least 0.9. Surface absorptivity must be at least 0.8. If significant, convective heat transfer must be considered on the basis of still air at 800°C.
Criticality	Criticality Safety	Waste package flooded and fully loaded with criticality design basis fuel except for one assembly that exceeds the design basis
		Waste package dry and fully loaded with fuel that exceeds the criticality design basis
		Waste package dry, fully loaded with criticality design basis fuel, with collapsed basket

only a common starting point for future design basis event consequence analyses and do not represent estimates of actual doses to individuals expected to result from any design basis events. The actual dose consequences resulting from a design basis event will be determined in future analyses. The method used for this analysis involves the following four steps:

1. Identify bounding fuel characteristics for commercial spent nuclear fuel. For pressurized and boiling water reactor spent nuclear fuel waste packages, this was the advanced conceptual design waste package thermal/shielding design basis fuel (Vol. III, Sect. 5 of CRWMS M&O, 1996b). For single assembly drop events, the bounding assembly was the highest burnup assembly (pressurized water reactor, 74.6 GWd/MTU) in the 1993 Energy Information Administration projections of future discharges. This step was unnecessary for defense high-level waste glass because only maximum values for radionuclide inventory per canister are available for this waste form.

2. Retrieve radionuclide inventories from the appropriate source for each waste form using the waste form characteristics identified in step 1. The characteristics data base was the source of radionuclide inventories for all but the 100 percent bounding pressurized water reactor assembly. Because the burnup of this assembly exceeded the upper limit of the characteristics data base, the SAS2H sequence of the SCALE 4.3 code was used to obtain the radionuclide inventory.

3. Apply U.S. Environmental Protection Agency (EPA) inhalation and submersion dose conversion factors (EPA, 1988) for the gonad, breast, lung, red marrow, bone surface, thyroid, remainder, and the whole body (effective dose) as appropriate to obtain the organ dose equivalent/unit waste form for each radionuclide.

4. Identify radionuclides to be used as the source term for each waste form. The nuclides included in the source term will be those that are the dominant contributors to 99.9 percent of the t[illegible] l dose per unit waste form for at least one organ. Those nuclides specified in N[illegible] acceptance criteria for accident analysis source terms for dry storage facilities (p. 7-7 of NRC, 1997b) will also be included regardless of their contribution to dose.

The results of this activity are reported in the Source Terms for Design Basis Event Analyses QAP 3-9 design analysis (CRWMS M&O, 1996v). This list is closely related to the repository design basis event list that appears in Section 4.2.1. Both lists were developed from the Preliminary MGDS Hazards Analysis, which is discussed in Section 4.2.1. The analyses summarized in this Section (5.1.3) considered all the events listed in Section 4.2.1 that were related to the waste package. Some additional waste package related events were also identified as a result of recent MGDS design changes. Events were then screened, as discussed above, based on credibility and ability to directly impact the waste package. Unlike the analysis described in Section 4.2.1, the waste package event analysis did not classify events as Category 1 or 2 because this analysis dealt only with frequencies of waste package-related initiating events.

frequencies of subsequent events in the radiological release sequence are required for categorization as defined in 10 CFR 60.

Subactivity 1.10.2.3.6 - Cost estimation. The purpose of this subactivity is to provide up-to-date cost estimates for the development and production of waste packages. The disposal container design and the materials selection have been substantially changed, and in support of this an update on cost was provided to support the Preliminary Draft Program Cost Estimate Report. This is an ongoing activity, and the cost estimates will be continually refined to support the viability assessment.

Forecast: The following design evaluation activities are forecast for the next reporting period:

- Thermal analyses in FY 1997 will continue to use three-dimensional models to analyze several waste packages. However, revised design basis fuels will be used to provide extra rigor to support the viability assessment. The three-dimensional models will be used to predict the temperatures of components of the engineered barrier segment outside the waste package. Additional thermal analyses of the interior of the waste package will be used to choose designs that will appropriately control internal temperatures.

- Structural evaluations will continue, including evaluations of design basis events for the waste package. These design analyses will evaluate the basket assembly structural strength against dynamic and static loads and containment barrier performance under impact loads. In addition, one work activity will focus on developing the engineered barrier segment waste package drip shield and additional barrier component designs to the level of detail required for the viability assessment.

- The major disposal criticality activity will be the issuance of Revision 1 of the Disposal Criticality Analysis Methodology Technical Report. Revision 1 will describe a further refined and developed criticality analysis methodology from that presented in Revision 0 of the report. The Project plans to issue an initial release of the Disposal Criticality Analysis Methodology Topical Report in FY 1998.

  The potential for external criticality caused by degradation and transport of highly enriched DOE-owned spent fuel will continue to be evaluated.

- Work on identifying design basis events will continue with the performance of multi-disciplinary analyses. The analyses will determine waste package response to design basis events.

- Cost estimation will continue as needed to reflect advances in and revisions to design. This work will eventually support the total system life cycle cost analysis for the 1998 viability assessment.

5.1.4 Design Activity 1.10.2.4 - Material Selection Design Support

The purpose of this activity is to perform analyses and make recommendations based on these analyses for material selection in support of engineered barrier design. Waste package material selection will significantly impact the ability of the waste package components to withstand degradation in the repository environment for an extended period of time. Materials for other components of the engineered barrier system must also be selected to promote performance.

Subactivity 1.10.2.4.1 - Materials selection process The purpose of this subactivity is to perform analyses to support materials selection for the waste package.

An analysis of waste package materials for viability assessment was discussed in Progress Report #15 (DOE, 1997e). Work during this reporting period has emphasized materials for the waste package supports and the invert materials rather than the waste packages themselves. Work on materials selection for these two components is in progress and will be reported in the next progress report.

An important source of uncertainty in waste package performance is uncertainty regarding the near-field environment. Efforts are being made to obtain a clearly defined environment in which the waste packages are expected to provide their functions of containment and controlled release.

Subactivity 1.10.2.4.2 - Container shell. The purpose of this subactivity is to determine the materials of choice for the waste package containment barriers. The barriers include a corrosion allowance barrier and a corrosion resistant barrier.

The material for the corrosion allowance barrier remains ASTM A 516 (carbon steel), and the material for the corrosion resistant material remains ASTM B 443 (Alloy 625). If corrosion testing shows that the corrosion resistance of ASTM B 443 is inadequate, other highly corrosion resistant materials will be considered as possible replacements.

Subactivity 1.10.2.4.3 - Shield plug. The purpose of this subactivity is to determine the materials for the waste package shield plug.

A shield plug is not included in the current design of the disposal container. Accordingly, no effort has been made in selecting shield plug materials. No work is forecast in this area.

Subactivity 1.10.2.4.4 - Spent nuclear fuel basket (structural/thermal). The purpose of this subactivity is to determine the structural materials for the basket to be contained inside the waste package. All containers for spent nuclear fuel incorporate basket assemblies. These assemblies provide structural support for the spent nuclear fuel assemblies, assist in heat transfer, and assist in criticality control. This section discusses those components for which criticality control is a secondary function.

The structural and thermal components include the basket tubes and the basket guides. For both of these components, ASTM A 516 remains the reference material. Other materials with higher thermal conductivity will be considered if analysis indicates that some spent fuel (such as fuel with exceptionally high burnups) requires this to control fuel temperatures. Disposal containers for boiling water reactor fuel do not include basket tubes.

Subactivity 1.10.2.4.5 - Spent nuclear fuel basket (criticality). The purpose of this subactivity is to determine the materials to be added to the basket to control criticality in the waste form. This section discusses those components for which criticality control is the primary function, although these components may also perform structural and thermal functions.

In current designs, the components important for criticality control take the form of slotted, interlocking plates. Past efforts in this subactivity began with surveying the corrosion behavior of candidate basket materials in the available literature and performing short term corrosion tests on them. The results of this work, together with a comparison of likely costs of the candidate materials, resulted in attention being focused on the boron-containing stainless steels and boron carbide as the leading candidates. More detailed studies were performed on the boron-containing stainless steels, including the synthesis of mixed metal borides in macroscopic sizes, matching the composition of the micron-scale dispersed borides in the stainless steels. These were used in electrochemical polarization experiments to determine their corrosion properties. Borides were found to have a higher open-circuit potential than the stainless steel matrix material when a composition near that of Type 304 stainless steel was used, indicating that the borides would be galvanically protected in this instance. Nevertheless, because greater overall durability could be obtained with little additional cost by choosing a grade of stainless steel similar to Type 316, the reference material selected was Neutronit A978. Neutronit A978 or equivalent remains the reference material for these plates. Neutronit A978 is a proprietary grade of stainless steel boron plate produced by Böhler Bleche GmbH of Mürzzuschlag, Austria. It is preferred over standard grades of stainless steel boron plate, such as those described by ASTM A 887, because the molybdenum content of Neutronit A978 is expected to provide better resistance to corrosion. Corrosion resistance is important in maintaining long-term criticality control. Borides have a lower open circuit potential than that of the stainless steel matrix, and the important property for long-term durability is the corrosion performance of the stainless steel matrix.

In this reporting period, sheet samples of Neutronit A978 were obtained from the manufacturer. The sheet was cut into corrosion coupons in preparation for placing the samples into the long-term corrosion test facility.

Boral has also been discussed as a potential material for criticality control in the waste package although there has been concern about its long-term resistance to degradation. Samples of anodized Boral have been received from the manufacturer, AAR Advanced Structures, Inc., for scoping corrosion tests. This additional material is being subjected to scoping tests at the request of Holtec International. Holtec International believes the anodizing treatment will give the Boral significantly improved performance over that observed for nonanodized Boral

Because of the recommendation of the National Academy of Sciences Committee on Yucca Mountain Standards that the EPA should consider times out to 1 million years in their new standard, interest has been drawn to the behavior of criticality control materials out to such times. Consequently, planning is underway to perform experiments that will provide information on the extent of retention of boron by the waste package corrosion products and provide a basis for predicting long-term performance.

Irradiation corrosion testing of boron carbide was not performed as planned during this reporting period. Preliminary discussions have begun with cobalt-60 facility personnel to design experiments that will not require extensive health and safety planning and approvals.

Subactivity 1.10.2.4.6 - Filler material. The purpose of this subactivity is to consider materials for a filler that might be added to the waste package to displace potential neutron moderators and help control criticality.

With current designs, disposal containers for uncanistered fuel are not expected to require filler material for criticality control. Disposal containers for canistered fuel may require filler material. However, because of the Project emphasis on uncanistered fuel, no additional work on filler material is planned.

Subactivity 1.10.2.4.7 - Fill gas. The purpose of this subactivity is to determine the appropriate choice of inert material for fill gas to be added to the waste package.

Helium remains the reference fill gas because it provides a good combination of inertness and high thermal conductivity.

Forecast: Work on materials selection will continue. Reference materials for the waste package supports and invert material will be formally selected during the coming reporting period. Structural work on basket materials will continue, with emphasis on corrosion testing of the A978 material, irradiation corrosion testing of boron carbide, and the interaction of boron with corrosion products.

5.1.5 Design Activity 1.10.2.5 - Performance Evaluations

This design activity includes work on materials performance. Issues addressed are container oxidation and corrosion, degradation by mechanical stress, and thermal degradation of fuel cladding.

Subactivity 1.10.2.5.1 - Container oxidation and corrosion. The purpose of this subactivity is to analyze degradation of potential waste package container materials as a result of oxidation and corrosion.

The rate of waste package degradation depends on the near-field hydrothermal and geochemical environment. Corrosion of the containment barriers, for example, will depend on the amount of water that contacts the waste package and the composition of the water. Progress on predictions of container oxidation and corrosion is currently hampered by the lack of a clearly defined environment. Efforts are in progress to develop a consensus on the near-field environment and produce controlled documentation of what environment should be used for design.

Subactivity 1.10.2.5.2 - Degradation of fuel cladding. The purpose of this subactivity is to analyze the degradation of spent nuclear fuel cladding. Fuel cladding may be useful as a barrier for controlling the release of radioactive materials to the environment, although credit has not been taken for the cladding in total system performance assessments conducted to date.

Work focused on two areas supporting the determination of whether credit can be taken for cladding in performance assessment. The first was relating the design of the cladding to degradation (CRWMS M&O, 1997t). For example, if dry oxidation or aqueous corrosion are important mechanisms for exposing fuel, the thickness of the cladding will be important. The intention is to define a design basis cladding for each of the significant degradation mechanisms. The design basis cladding would then be used with degradation models to determine the amount of performance that could be claimed for cladding. The results of the study are comparable to earlier informal estimates, but this study provides much clearer documentation of how the various values were obtained. Because the results indicate that the earlier estimates were fairly accurate, there is still reason to believe that cladding will provide significant control of releases.

It is not possible to construct a rigorous description of the spent nuclear fuel that will be placed in a high-level radioactive waste repository. Nuclear reactor licensees generally characterize only a few spent fuel assemblies, so there are limited data even for existing spent nuclear fuel. Predicting the characteristics of future spent nuclear fuel is even more difficult. Nuclear reactors will presumably continue to operate and produce spent nuclear fuel, but much of this fuel has not been manufactured yet, and designs and materials may change in the future. Operating conditions may also change, so the amount of fuel degradation at the time of reactor discharge is also uncertain. Finally, spent nuclear fuel may be stored under a variety of conditions until a repository is constructed and begins operation, and the amount of damage that will occur during storage is difficult to predict.

Commercial light-water reactor spent fuel in the United States may be divided into two classes: that with stainless steel cladding and that with zirconium alloy cladding. In developing a design basis cladding for spent nuclear fuel, it is important to determine what quantity of fuel falls into each class, because the composition differences between stainless steel and zirconium alloy cladding are expected to produce large differences in corrosion performance after waste package breach. From data in DOE reports and data bases, plus information from nuclear plant staff, the fuel with stainless steel cladding contains 723 metric tons of uranium. Of this, about 8 percent is clad with Type 348H stainless steel; the balance is clad with Type 304 stainless steel. Since stainless steels are much less corrosion resistant than the zirconium alloys, no additional consideration was given to taking credit for the stainless steel clad.

To date, five degradation mechanisms for zirconium alloy clad fuel have been considered: creep rupture, dry cladding oxidation, dry fuel oxidation, aqueous cladding corrosion, and external mechanical loading. Additional mechanisms may be considered in the future.

Creep rupture is driven by stress in the cladding that results from fill gas and released fission gas inside the fuel rods. The stress in the cladding is the gas pressure times a ratio that depends on the diameter and wall thickness of the cladding. For pressurized water reactor fuel, without correction for corrosion, the ratio of circumferential stress to pressure can be as high as 8.9. For boiling water reactor fuel, the highest ratio of circumferential stress to pressure is 8.3.

In contrast to creep rupture, dry cladding oxidation and aqueous cladding corrosion would cause exposure of fuel by consuming the thickness of the cladding. Several types of pressurized water reactor fuel have been found with a cladding thickness of 0.0225 in. (0.572 mm). Almost all boiling water reactor fuel has a cladding thickness of at least 0.030 in. (0.762 mm), but there are eight assemblies (not assembly types) with thinner cladding.

Cladding failure by fuel oxidation requires an existing breach in the cladding. Accordingly, the rate of failure does not depend on the cladding design.

If the containment barriers are badly degraded, the waste package may expose the fuel assemblies to external mechanical loads. Sources of mechanical loads include the products of corrosion of the containment barriers and rubble from the crown of the drift. External loading is expected to be much more severe for pressurized water reactor fuel than for boiling water reactor fuel because boiling water reactor fuel rods are normally enclosed in and protected by flow channels, and it is assumed these channels will be disposed of with the fuel rods.

It is difficult to quantitatively describe the loading on the fuel rods. The loading on the fuel assemblies may be either static or dynamic. The loads may be imposed by large or small pieces of rubble. The fuel assembly can be either intact or degraded. A conceptual model has been developed in which the fuel rods act as horizontal beams with distributed loads, supported intermittently by the spacer grids. The rods are assumed sufficiently stiff that those in the top layer do not sag into the layer below. Under this assumption, the top layer of rods supports the entire rubble bed; rods in lower layers do not share the load. Such a model might be appropriate if the rubble is substantially smaller than the distance from one spacer grid to the next but larger than the spaces between adjacent rods. Such rubble might be provided by backfill or rubble from host rock with closely spaced joints. Because the loading configuration is uncertain, the model is given simply as an example, not necessarily as a conservative or realistic description of the loads that would occur in a repository.

In this model, the stress in the cladding depends on four quantities: the outside fuel rod diameter, the cladding wall thickness, the fuel rod pitch, and the distance between spacer grids. The maximum stress in the cladding is the product of the area-averaged pressure of the rubble bed and a stress multiplication factor, which is a function of the four quantities just listed. One pressurized water reactor assembly type has a stress multiplication factor of 13,500, and several additional assembly types have stress multiplication factors of over 11,000. Because the area-averaged pressure of the rubble bed does not depend on the assembly design, the fuel assembly types with the largest stress multiplication factors are expected to suffer the most damage by external loading. Stress multiplication factors were not calculated for boiling water reactor fuel because the fuel rods will normally be protected by the flow channels.

The second area of focus for this activity was the effects of zirconium-zinc interactions (CRWMS M&O, 1996w). This work was motivated by a hydrogen burn in a dry storage cask at the Point Beach nuclear power plant on May 28, 1996. The hydrogen was apparently generated by corrosion of metallic zinc in a paint that was used to control corrosion on the interior surfaces of the cask. The zinc corroded, producing hydrogen, while the cask was submerged in the fuel pool, and the hydrogen was ignited during welding of a shield lid. As a result of the hydrogen burn, there has been renewed scrutiny of the use of zinc in dry storage casks for spent nuclear fuel. It has been postulated that, during storage, zinc vapor could attack and degrade the cladding. This question has implications for fuel disposal as well, because fuel damaged during dry storage would be expected to have poorer performance after disposal.

It was determined that zinc could be transported as a vapor to the fuel cladding, but the amount of zinc available is only sufficient to affect a small fraction of the zirconium in a cask. However, there may be nonuniformities in the reaction. For example, attack may be more severe at grain boundaries or in areas of unusually high or low temperature. Nonuniform attack could result in greater local cladding degradation. No specific evidence was found that zinc does or does not degrade the mechanical properties of zirconium, but several qualitative arguments indicated that significant degradation will not occur. Thus, results of work in this area are not conclusive. Contact will be maintained with users of dry storage devices that contain zinc to collect additional information on the subject should it become available.

<u>Forecast</u>: An analysis of cladding designs and their effects on degradation will be documented. This document will serve as an input to total system performance assessment models.

5.2 INTERACTION BETWEEN THE WASTE PACKAGE AND THE POSTEMPLACEMENT NEAR-FIELD ENVIRONMENT (SCP SECTION 8.3.4.2)

5.2.1 <u>Design Activity 1.10.1.1 - Consideration of 10 CFR 60.135(a) Factors</u>

The purpose of this activity is to explicitly show that the factors specified in 10 CFR 60.135(a) for interactions between the waste package and its environment have been considered in waste package design. These factors include the in situ physical, chemical, and nuclear properties of the waste package and the effect of processes such as solubility, oxidation, corrosion, and hydriding, etc.

This activity mostly involves compiling information obtained in other activities. Therefore, the discussion below refers to accomplishments in the other activities.

Emplacement-Scale Thermal Analyses

The first step in the thermal evaluation of the waste package is to determine the time-dependent response of the repository to the decay heat of the emplaced waste packages. This emplacement-scale evaluation must consider that the waste package both affects (through thermal loading) and is affected by the conditions of its near-field environment. Although a given thermal (mass) loading is typically characterized by a single number, such as 83 MTU/acre, the thermal response of the repository depends on the heat generation as a function of time in the various waste packages, which have different values of this important process variable. The heat generation in turn depends on the characteristics of the waste stream such as spent fuel age, receipt rates, delivery scenarios (youngest-fuel-first versus oldest-fuel-first), waste package size, emplacement spacing, and design basis fuel.

As reported in Progress Report #15, emplacement-scale evaluations have been performed to support systems studies considering both emplacement drift backfill and thermal loading and to advance the waste package design effort. These evaluations indicate drift wall temperatures largely depend on the assumptions used to estimate the average waste stream heat loads and on the variability in heat loads within the waste stream.

During this reporting period, further emplacement-scale evaluations were performed to determine the impact of higher thermal loadings (100 MTU/acre) and selective waste package placement. This work, reported in Section 5.1.3 (Activity 1.10.2.3.1) of this progress report, concluded that thermal loadings significantly higher than the 83 MTU/acre assumed in the advanced conceptual design would cause peak drift wall temperatures to approach or exceed limits.

Evaluations continued of the "line loading" concept, in which waste packages would be placed close together to achieve a more uniform temperature distribution along the drift than would be reached using the advanced conceptual design waste package spacing as discussed in Section 5.1.3. The evaluation found that, by separating waste packages containing the hotter design basis fuel, peak drift wall temperatures near the hottest package were lowered by nearly 10°C. Estimates of drift wall temperatures, both peak and average, remain significantly above the thermal goal, and temperature remains the primary detraction to the feasibility of the line loading concept.

Waste Package Scale Thermal Analysis

Specific finite element models of waste packages for uncanistered fuel have been developed to support waste package-scale thermal analyses of potential design changes. Evaluations this reporting period, reported in Section 5.1.3, considered support guides and the potential thermal shunts, as well as different materials. The results of this work supported, from a thermal performance standpoint, use of aluminum alloy thermal shunts in the basket.

Seismic Factors

To examine seismic effects, a seismic factor of 1.66 was applied to the load of a static finite-element analysis of a three-dimensional half-symmetry finite-element model of the proposed waste package support structure. A preliminary result for the seismic design of the support and pier structure was obtained. The results were compared as discussed in Section 5.1.3 of this progress report, to the material yield strengths to determine locations of any permanent deformations in the system. The resulting maximum stresses in the support structure were less than the yield strength of the materials. Therefore, the structural performance of the waste package support system components were considered acceptable under seismic loading.

Container oxidation and corrosion

The rate of waste package degradation depends on the near-field hydrothermal and geochemical environment. Corrosion of the containment barriers, for example, would depend on the amount of water that contacts the waste package and the composition of the water. Progress on predictions of container oxidation and corrosion awaits clearer definition of the near-field environment. Efforts are in progress to develop a consensus on the near-field environment and produce controlled documentation of what environment should be used for design. Near-field environment work is reported in Sections 5.2.2 through 5.2.7 of this progress report. Work on container oxidation and corrosion is also discussed in Section 6.9 of this progress report.

Degradation of fuel cladding

Work was performed to better understand cladding degradation resulting from interactions with the environment. This work is discussed in Section 5.1.5, Activity 1.10.2.5.2, of this progress report. For example, if dry oxidation or aqueous corrosion are important mechanisms for exposing fuel, the thickness of the cladding will be important. The intention of this ongoing work is to define a design basis cladding for each of the significant degradation mechanisms. The design basis cladding would then be used with degradation models to determine the amount of performance that could be claimed for cladding. The results of the study continue to provide reason to believe that cladding will significantly control the release of radionuclides from spent fuel waste.

Forecast: Modeling work on 10 CFR 60.135(a) factors will focus on developing the ability to more accurately account for (a) spatially variable ambient percolation flux distributions, (b) fracture-matrix interaction, and (c) spatially variable natural system properties such as bulk permeability in the drift-scale and hybrid drift-scale–mountain-scale thermal-hydrological models. A major goal of this effort is to develop the ability to more accurately predict seepage flux distributions along emplacement drifts. The thermal-hydrological modeling effort will closely collaborate with site-scale thermal-hydrological modeling activities and with altered zone activities. There will also be increased effort in modeling drift-scale thermal-hydrological behavior to assist in model abstractions required by total system performance assessment and viability assessment activities.

Work in the area of fuel cladding degradation will include documenting an analysis of cladding designs and their effects on degradation. This document will serve as an input to total system performance assessment models.

5.2.2 Study 1.10.4.1 - Characterize Chemical and Mineralogical Changes in the Postemplacement Environment

The purpose of this activity is to establish, to the degree required in Performance Issues 1.4 and 1.5 (SCP Sections 8.3.5.9 and 8.3.5.10), the information necessary to characterize the chemical and mineralogical properties and processes of the waste package environment for anticipated and certain unanticipated conditions. To accomplish this objective, the study will determine the effects of chemical reactions on rock-water systems of the repository horizon over a range of temperatures and chemical conditions that bound the postclosure waste package environment.

This study had seven activities in the SCP (Activities 1.10.4.1.1 through 1.10.4.1.7). Before writing the study plan, the study was divided into two studies (near-field geochemistry and introduced materials). When the near-field geochemistry study plan was drafted, the remaining material was organized differently than in the SCP. To avoid confusion between similarly numbered "old" and "new" activities, the "new" activities were renumbered with higher numbers. The cross linkage of the original activities to the current activities is included in Appendix A of this progress report The status of the current activities is provided below.

The only funded activity in FY 1997 was completion of a report on bounds of water chemistry that may contact emplaced materials (Glassley, 1997). The report summarized results obtained to date and, along with the Near-Field and Altered-Zone Environment Report (Wilder, 1996) published last year, covered the activities described below, none of which are funded for this fiscal year. The report concludes that water, interacting with repository materials, will be a mixture of ambient waters and condensate that have interacted with rock and fracture mineralogy, and the residues of evaporative processes. The largest volume is expected to travel via fracture flow. The range of compositions that may be expected will fall within the following bounds:

Bounding Condition #1 - The water will have compositions completely dominated by evaporative processes, in which case the composition will depend on the degree of evaporation.

Bounding Condition #2 - The water is dominated by condensate, in which case the actual composition will depend upon the extent of interaction of water with minerals along the pathway. This, in turn, is a function of flow path length, and fluid velocity. It is expected that this water will be a significant, if not dominant, component of water interacting with repository materials.

Bounding Condition #3 - The water is dominated by fracture flow of ambient fluids percolating through the repository block.

Activity 1.10.4.1.8 - Hydrothermal testing of vitric and tuffaceous rocks under saturated conditions. The objective of this activity is to conduct a series of long-term saturated tests to determine the chemical characteristics of the water and the solid phase reaction products that may develop during the interaction of rocks near the potential repository horizon, with reference ground water and other waters at elevated temperatures. Results from these tests are particularly applicable to the deeper, saturated units or to unsaturated units under specific conditions wherein condensate exposes the host rock to above-ambient water availability.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.9 - Hydrothermal testing of vitric and tuffaceous rocks under unsaturated conditions. The objective of this activity is to conduct a series of long-term tests similar to those of Activity 1.10.4.1.8, except that water activity will be controlled to ensure that the activity is always less than 1.0. These tests will evaluate how pore water chemistry and secondary mineralogy may evolve under conditions where water activity is less than 1.0. Reaction rates and mechanisms may also be substantially changed under these conditions. Furthermore, the degree of hydration may change for hydrous phases with a corresponding change in mineral volume. This work is designed to complement other work addressing mineral stability and geochemical evolution of the site (Studies 8.3.1.3.3.2 and 8.3.1.3.3.3, Sections 3.2.5 and 3.2.6 of this progress report).

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.10 - Mineral dissolution and precipitation. The objective of this activity is to obtain knowledge of the dissolution kinetics of the phases present in the host rock of the near-field environment and the precipitation kinetics of product mineral phases. This information is required to interpret observed changes in fluid composition and associated development of product mineral phases in hydrothermal rock-water interaction studies.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.11 - Ion exchange and sorption. The objective of this activity is to obtain knowledge of the effect that ion exchange and sorption may have on the composition of mineral phases and the composition of coexisting water. This information is required to interpret observed changes in fluid composition and associated development of product mineral phases in hydrothermal rock-water interaction studies.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.12 - Rock-water interaction and water chemistry changes in the presence of a radiation field. The objective of this activity is to obtain knowledge of the interaction of ionizing gamma radiation with the air-steam atmosphere and pore water in the near-field environment, with the concomitant spectrum of possible effects on the rock, pore water, and emplaced materials.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.13 - Simulation of rock-water interaction. The objective of this activity is to conduct simulations of the rock-water interaction experiments described in Activities 1.10.4.1.8 and 1.10.4.1.9, using as appropriate, models and data generated from Activities 1.10.4.1.10, 1.10.4.1.11, and 1.10.4.1.12. The activity also simulates natural systems in which processes of interest occur. The activity evaluates computer codes and data bases, but does not actually develop codes or data bases. The results of this activity will allow simulations of repository conditions thousands of years into the future.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.14 - Validation of EQ3/6 reaction path modeling codes. The objective of this activity is to validate the EQ3/6 code package to be used in Activity 1.10.4.1.13. This activity will use laboratory hydrothermal experiments not used in previous modeling efforts, analogous natural systems, field-based studies, and ESF studies to validate the calculational approach to reaction path modeling.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.15 - Rock-water interaction simulation of scenarios for license application. The objective of this activity is to use the EQ3/6 code to simulate rock-water interactions for short- and long-term periods, for specific scenarios required for license application. The results will establish the geochemical and mineralogical characteristics of the waste package environment for expected and certain unexpected conditions. The characteristics will include the expected changes in primary and secondary mineralogy that would occur as a result of the interaction of the vadose water with the waste package environment thermal and radiation fields, and with the host rock. The compositional evolution of the vadose water will also be established for the range of temperatures and radiation doses expected in the waste package environment.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.1.16 - Experiments and simulations to determine the effect of geochemical processes on hydrological processes. The objective of this activity is to determine, through experiments, simulations, and study of natural systems, how geochemical processes couple with hydrological processes under the expected post-emplacement hydrothermal conditions. The geochemical processes include dissolution and precipitation of minerals. The hydrological properties of fracture apertures, pore sizes, pore and fracture connectivity, and imbibition properties of the rock will in turn modify the flow pathways and flow rates of water and vapor as heating and cooling of the repository occur. This activity will evaluate the extent to which chemical changes will modify the hydrological properties and will also determine under what conditions these changes are of greatest significance for geochemistry.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: A summary report will be produced of the information on the effect of rock-water interaction on water chemistry. This information will be used to determine changes in the local hydrogeological properties of the rock. Dissolution of pre-existing materials could increase local porosity and permeability, while precipitation of minerals could decrease porosity and permeability. Porosity and permeability are important parameters affecting repository performance, so this work will be an input to future repository performance assessments.

5.2.3 Study 1.10.4.2 - Hydrologic Properties of Waste Package Environment

The objectives of this study are to conduct experimental and modeling studies relevant to the range of potential thermal loads to

1. Identify hydrological and transport processes at Yucca Mountain that significantly affect waste package performance, and radionuclide release and transport

2. Develop a detailed conceptual and quantitative understanding of decay-heat-driven flow processes that govern the waste package environment, including temperature, relative humidity, and flow conditions throughout the repository and the engineered barrier system

3. Conduct experiments and develop related models to assess the impact of decay-heat-altered matrix and fracture properties on nonequilibrium fracture flow

4. Develop and conduct laboratory and in situ tests for model validation and hypothesis testing that provide the basis for confidence building for coupled thermal-hydrological-geomechanical-geochemical process models required for total system performance assessment.

This study had three activities in the SCP (Activities 1.10.4.2.1 through 1.10.4.2.3). When the study plan was written and subsequently revised, the material was organized differently than in the SCP. To avoid confusion between similarly numbered "old" and "new" activities, the "new" activities were renumbered with higher numbers. The cross linkage of the original activities to the current activities is included in Appendix A of this progress report. The status of the current activities is provided below.

Work during this reporting period regarding Study 1.10.4.2 was confined to Activity 1.10.4.2.6 (model development and analysis of thermal-hydrological flow and transport).

Activity 1.10.4.2.4 - Laboratory hydrological property measurements. The objectives of this activity are to determine the hydrological properties of repository horizon Topopah Spring Tuff samples and other rock units that may fall within the altered zone. The properties include effective porosity, saturated liquid- and gas-phase permeability, matrix suction potential vs. liquid saturation, the effective coefficient for the binary diffusion of air and water vapor, and the Klinkenberg coefficient. The properties will be measured under ambient conditions and thermally altered conditions that are relevant to the heating and cooling cycle for a range of potential thermal loads.

No progress was made during this reporting period; this was an unfunded activity.

Activity 1.10.4.2.5 - Model validation experiments. The objectives of this activity are to develop and conduct laboratory tests for model validation and hypothesis testing that provide the basis for confidence building for coupled thermal-hydrological-geomechanical-geochemical process models required for total system performance assessment. The experiments will test the adequacy of the models to represent coupled thermal-hydrological-geomechanical-geochemical flow and transport processes at Yucca Mountain that significantly affect waste package performance, radionuclide release, and radionuclide transport.

No progress was made during this reporting period; this was an unfunded activity.

Activity 1.10.4.2.6 - Model development and analysis of thermal-hydrological flow and transport. The objectives of this activity are to conduct modeling studies for the range of potential repository thermal loading options and for the thermal loading cycle to

1. Identify coupled thermal-hydrological-geomechanical-geochemical processes, transport processes, and ambient site conditions (e.g., bulk permeability distribution) that significantly affect the waste package and engineered barrier system environment. In other wrods, those processes and conditions that affect waste package performance and radionuclide dissolution, release, and transport will be identified. Their effect on temperature, relative humidity, and flow conditions throughout the repository and engineered barrier system will be emphasized.

2. Determine the parameter sensitivity of the thermal-hydrological behavior in the near field and engineered barrier system to a range of expected site conditions, waste package designs, repository configurations, waste package loading scenarios, and repository operational options (e.g., ventilation and backfill). Of particular importance are dryout and rewetting behavior.

3. Develop mathematical and numerical models of repository-heat-driven flow and radionuclide transport, emphasizing the waste package and engineered barrier system environment. These models should be capable of treating the importance of coupled thermal-hydrological-geomechanical-geochemical and transport processes in the thermally altered zone. These processes include (a) nonequilibrium fracture-matrix interaction, (b) coupled reactive transport, (c) the effects of coupled fracture-aperture

deformation, and (d) the effects of heat-altered thermal and hydrological flow and transport properties.

4. Establish hypotheses critical to predicting thermal-hydrological behavior and engineered barrier system performance. Laboratory- and field-scale experiments that critically test these hypotheses will be designed.

5. Develop validated subsystem models of the waste package and engineered barrier system environment. The experiments of Activity 1.10.4.2.5 will be used, in part, to validate the models.

The following paragraphs summarize progress made this reporting period on this activity.

Near-Field and Altered-Zone Thermal-Hydrology

Modeling of near-field thermal-hydrology continued, using a combination of mountain-scale and drift-scale models and the NUFT flow and transport code.

Three-Dimensional Thermal-hydrological Drift-Scale Model Analyses and Revisions

Much of the work continued to be conducted with the three-dimensional thermal-hydrological drift-scale model that explicitly represents six different waste package types, resulting in a waste package inventory that is representative of that assumed for the Advanced Conceptual Design (CRWMS M&O, 1996b). The model has been modified to reflect the evolving assumptions about waste package and drift sizes and waste package and drift spacings. A benchmark study was also conducted with a four-waste package three-dimensional drift-scale conduction-only NUFT-based model. The results of this study were compared with those obtained with an ANSYS-based conduction-only model developed by the Project. Averaged over the four waste package locations, both models predicted almost identical drift-wall and waste package surface temperatures, while the NUFT-based model predicted greater axial temperature variability, with higher temperatures for the hottest waste package location and cooler temperatures for the coolest waste package location.

Both of the conduction-only models represent how thermal radiation distributes the decay heat from the waste packages to the surfaces of the emplacement drift (e.g., drift wall). The variability of drift-wall and waste-package temperatures (as a function of axial position along the emplacement drift) is particularly sensitive to the representation of thermal radiation within the drift. Thermal homogenization of temperatures along the drift is most readily achieved via radiative heat transfer from (1) waste package to drift surfaces, (2) drift surface to drift surface, and (3) waste package to waste package. The third means of heat transfer is only significant if the waste packages are sufficiently close to each other, as is the case with the line-load design. Heat transfer in the rock (by conduction or convection) plays a minor role in homogenizing the axial temperature distribution along the drift.

The major purpose of the comparison between the NUFT-based and ANSYS-based conduction-only models was to compare how thermal radiation influences the axial distribution of drift-wall and waste package along the drift. Because the NUFT-based model neglected drift-surface-to-drift-surface thermal radiation, it predicted somewhat less axial thermal homogenization than the ANSYS-based model. In more recent NUFT-based models (such as those used to analyze the drift-scale thermal test) drift-surface-to-drift-surface thermal radiation has been included. This has been found to be important to fully account for the influence of thermal radiation on the distribution of heat flux along the drift-wall surfaces.

<u>New Drift Seepage Model</u>

A new three-dimensional NUFT-based drift-seepage model was developed that uses extremely fine gridding (gridblocks roughly 10 cm on a side) and that can precisely represent the cylindrical geometry of the emplacement drift and waste package. This model can represent highly heterogeneous distributions of thermal and hydrological properties in the fractures and matrix, either using the Effective Continuum Model or Dual Permeability Model. The NUFT code has been enhanced to include a stochastic-field generation capability with the following attributes:

- Subdomains can be specified, each with its own statistically consistent stochastic field. For example, each lithological unit can have its own statistical properties.
- Any set thermal or hydrological parameters within a subdomain can be a function of up to three independent stochastic fields.
- The stochastic field generation works with the Dual Permeability Model and nested-mesh capabilities of the NUFT code.
- The stochastic field generation algorithm produces a spatially correlated normal or log-normal field for gaussian, exponential, or fractal distributions.

A parameter sensitivity study was conducted with the drift-seepage model to investigate the relationship between drift seepage flux (into the drift) and percolation flux for both homogeneous and heterogeneous conditions. Key parameters affecting the predicted threshold percolation flux for seepage into the drift were the van Genuchten beta parameter for the fractures and the heterogeneity of the major fracture flow paths. The beta parameter for the fracture is an indication of how well sorted the fracture aperture size distribution is. Small values of beta indicate a wide distribution of aperture sizes, while large values indicate that the aperture distribution is more uniform (as in a parallel plate). Increasing the beta value reduces the threshold percolation flux at which water is predicted to be able to seep into the drift. Increasing the heterogeneity in the fracture properties also is predicted to reduce the threshold percolation flux at which water is able to seep into the drift. For example, seepage into the drift would not be predicted to occur until about 200 mm/yr for a two-dimensional homogeneous case, while for a three-dimensional heterogeneous case, seepage into the drift would start to occur at around

50 mm/yr. Increasing the fracture beta parameter from 1.47 to 4.23 for the homogeneous case reduced the predicted threshold percolation flux from 200 to 10 mm/yr.

Pre-Test Analyses Of The Drift-Scale Thermal Test

Pre-test analyses of the drift-scale thermal test in the ESF were conducted with three-dimensional NUFT-based models. Many of the calculations were conducted with a conduction-only model that represents the influence of ventilation in all the mined openings in the thermal test area. This model explicitly represents heat flow from nine in-drift heaters (simulating waste packages), including radiation from the heaters to the drift surfaces. This model was used for design analysis to determine (a) the maximum expected temperature rise at selected locations in the therm ' .est area, (b) the ventilation requirements in the neighboring drifts, and (c) the insulation requirements for the thermal bulkhead that separates the heated and unheated portions of the heater drift. This information was provided to the ESF test and design organizations for use in design and construction of the test.

A sensitivity study of the influence of percolation flux on temperatures in the drift-scale thermal test was also conducted with a two-dimensional thermal-hydrological model of the drift-scale thermal test. For the 5-mm/yr case, the maximum drift-wall temperature at the center of the heater drift was more than 100°C lower than for the 0.05-mm/yr case. For the 5-mm/yr case the vertical dryout zone thickness was only half as large as in the 0.05-mm/yr case. The sensitivity study also compared the results obtained with a conduction-only version of the two-dimensional drift-scale thermal test model with those obtained with the thermal-hydrological models. The two-dimensional conduction-only model predicted a maximum drift-wall temperature that is more than 150°C higher than predicted by the two-dimensional thermal-hydrological model for the 5-mm/yr case. The conduction-only model predicted a maximum drift-wall temperature that is 50°C higher than predicted by the thermal-hydrological model for the 0.05-mm/yr case. Therefore, depending on the magnitude of percolation flux, the conduction-only model can be quite conservative with respect to predicting maximum drift-scale thermal test temperatures. Depending on how the models partition between matrix and fracture flow, there can be a large sensitivity of temperature to percolation flux. Though this sensitivity study was based on 0.05 mm/yr and 5 mm/yr, the results are expected to be a valid prediction of trends in temperature and dryouts versus percolation flux that would prevail at fluxes greater than 5 mm/yr.

Temperature and liquid saturation measurements made during the drift-scale thermal test will be highly indicative of the prevalent percolation flux in the thermal test area. Therefore, in addition to providing valuable information about coupled thermal-hydrological-geomechanical-geochemical processes, the drift-scale thermal test will provide a very useful means of determining the percolation flux conditions during the course of the test.

Modeling Study of the Single-heater Test

The single-heater test was modeled with a three-dimensional thermal-hydrological NUFT-based model that represented the effect of heat and mass transfer with all three ventilated drifts surrounding the test area. A sensitivity study of bulk permeability was conducted with this model. The dryout zone volume was found to increase with increasing bulk permeability, while the temperatures inside the boiling and superheated zones decrease with increasing bulk permeability.

Drift-scale Thermal-Hydrological Modeling and Abstraction

A major objective of the modeling and analysis tasks in near-field and altered-zone thermal hydrology is to provide a detailed description of "drift-scale" thermal-hydrological conditions in emplacement drifts as a function of time and location within the repository. This drift-scale thermal-hydrological description will be provided as response surfaces to support the total system performance assessment for the viability assessment. This description requires a model (or model-abstraction equivalent) that can represent three-dimensional mountain-scale thermal-hydrological behavior and three-dimensional drift-scale thermal-hydrological behavior. Calculating the drift-scale thermal-hydrologic conditions explicitly with the monolithic process-level thermal-hydrological model would require approximately 30 million gridblocks to achieve the same level of detail as in the drift-scale models used in Chapter 1 of the Near-Field and Altered-Zone Environment Report (Buscheck, 1996). Using a thermal-hydrological model with a grid-block density comparable to the drift-scale seepage model would require approximately 30 billion gridblocks. Consequently, it would be impossible to conduct a model calculation for just one realization, much less the thousands of realizations expected to be required for the total system performance assessment for the viability assessment. Therefore, a model-abstraction methodology has been developed that facilitates the analysis of thousands of realizations of the waste isolation system, thereby providing a calculational tool that can address variability and uncertainty in the distribution of natural system properties and conditions for a range of alternative repository and engineered barrier system designs.

The model-abstraction procedure for drift-scale thermal-hydrological conditions was developed during this reporting period. An outline of drift-scale and mountain-scale thermal-hydrological model calculations required by this model-abstraction methodology was also developed. Implementation of the model-abstraction procedure will occur during the next reporting period.

The primary objective of the drift-scale thermal-hydrological model-abstraction methodology is to determine the distribution of near-field environmental conditions that govern waste package degradation, waste-form dissolution, and radionuclide release from waste packages and the engineered barrier system, including the following:

- Temperature, relative humidity, liquid-saturation, air mass fraction in the gas phase, and liquid-phase flux at the drift-wall surface
- Temperature, relative humidity, air mass fraction in the gas phase, and liquid-phase flux on waste packages
- Temperature at the waste package centerline

- Liquid-phase flux that seeps into the drift
- Liquid-phase flux that reaches the drip shield (if present)
- Liquid-phase flux that drips onto waste packages
- Liquid-phase flux that drains into the invert.

This information will be provided as a function of the following:

- Waste package location and drift location
- Waste package type (waste form type; spent nuclear fuel age, burnup, and enrichment; MTU content)
- Waste package sequencing (i.e., arrangement of different waste package types in drifts)
- Thermal and hydrological property distributions
- Percolation flux (including magnitude and distribution)

The drift-scale abstraction approach involves superposition, using the results of complementary (or parallel) thermal-hydrological or thermal (only) models, including mountain-scale models, hybrid drift-mountain-scale models, and drift-scale models. The superposition process accounts for both three-dimensional mountain-scale thermal hydrological behavior and three-dimensional drift-scale behavior, including the following:

- Hydrostratigraphic layering, surface topography, and fault zones represented by the three-dimensional site-scale unsaturated zone flow model
- Conductive and convective heat transfer in the saturated zone
- The influence of potential heat sinks such as the central exhaust drift and the east and west service mains
- Alternative repository designs, waste package layouts, and thermal management approaches, such as the point- and line-load designs and the influence of lag storage and ventilation
- Alternative engineered barrier system enhancements, such as backfill and drip shields
- The influence of rockfall into the emplacement drift (if applicable).

The superposition process provides for computational efficiency and modularity, which facilitate the consideration of many variables and factors, including how the variability and uncertainty of thermal-hydrological-property and percolation-flux distributions influence thermal-hydrological behavior. The modularity of this approach allows for the ongoing development, modification, and refinement of a data base of results from process-level submodels that feed the drift-scale model-abstraction tool. Another motivation for this approach is to provide an abstraction framework that will more fully use thermal-hydrological model results from a variety of modeling groups in a unified and consistent fashion.

Forecast: Pre-test analysis of the drift-scale thermal test will continue with a three-dimensional thermal-hydrological NUFT-based model. Analyses of the single-heater test and large-block test will also be conducted with three-dimensional thermal-hydrological NUFT-based models. Calculations will be conducted with homogeneous and stochastic fields of properties, using both the Effective Continuum Model and Dual Permeability Model.

The drift-scale model-abstraction tool will be developed and delivered to those analysts developing the total system performance assessment for the viability assessment. A base of mountain-scale, drift-scale, and drift-seepage model results that feed the abstraction tool will continue to be developed during this period and also delivered to those developing the total system performance assessment for the viability assessment. The drift-scale model-abstraction tool and process-model data base will be augmented to address alternative repository and engineered barrier system designs and various waste-stream management options to support the repository design analysis effort.

The validity of the drift-scale model-abstraction tool will be tested against hybrid drift/mountain-scale models that explicitly represent the thermal-hydrological behavior.

5.2.4 Study 1.10.4.3 - Characterization of the Geomechanical Attributes of the Waste Package Environment

The objective of this task is to characterize the geomechanical response of the rock in the near field to the changing conditions expected to occur over the lifetime of the repository. This includes providing data from laboratory, field, and modeling investigations that can be used to support evaluations of the suitability of the site and to support licensing. Particular emphasis is on coupled processes and behavior at elevated temperatures and at long times.

This study had one activity in the SCP. When the study plan was written, the material was reorganized into three activities. The description of the original activity is included in Appendix A of this progress report. The status of the current activities is provided below.

Activity 1.10.4.3.1 - Block stability analysis. The objective of this activity is to identify and understand how the transport, physical, and mechanical properties of the near-field region are affected by thermal-mechanical behavior over time. This includes developing and applying constitutive models and numerical codes for analysis of geomechanical influences on the behavior of rock in the near-field environment over time. Analytic and numerical methods include continuum and statistical methods. Particular emphasis will be on the development and implementation of techniques for coupling of geomechanical with geochemical and hydrologic codes. Emphasis will also be placed on modification of existing codes as necessary to enhance the analysis of the field testing activities, and if appropriate, the integration of the geomechanical behavior into the performance assessment modeling of the site.

A study was initiated to estimate the bounds on the changes in fracture permeability from thermal-mechanical processes associated with the excavation of drifts and the emplacement of waste. Crucial to the site suitability evaluation is understanding the hydrologic response of fluids present in the proposed repository horizon to the development of a repository and the subsequent storage of high-level radioactive waste. Moreover, one of the inherent properties of rock that controls moisture movement and fluid flow is its permeability, and the permeability of rock is known to depend on stress and temperature. Furthermore, the stress field in the rock surrounding the drifts will be altered by both the excavation of drifts and the heating of the rock associated with waste emplacement and storage. Thus, the hydrologic behavior of rock surrounding emplacement drifts depends on the mechanical response of the rock to excavation and waste emplacement. In addition, the proposed repository horizon at Yucca Mountain contains a significant number of fractures, and the mechanical and hydrologic properties of fractured rock are not well understood. Prior work has shown that increasing stress across fractures causes a reduction in fracture aperture, and flow in a fracture can be related to approximately the cube of the fracture aperture. Generally, as compressive stress across a fracture is increased, the aperture is reduced, which reduces the fluid flow. More recent work indicates increases in shear stress across a fracture may also reduce the fracture permeability. Finally, while a preliminary understanding of flow in single fractures is now available, it is also widely accepted that the hydrologic behavior of a fractured rock mass is controlled by a few, well connected fractures in the rock mass.

Given this background, a methodology is now being developed to estimate bounds on the changes in fracture permeability from thermal-mechanical processes associated with the excavation of drifts and the emplacement of waste. A three-step procedure has been developed to estimate permeability caused by construction-induced stress changes and by heating. First, a numerical stress model (FLAC or ABAQUS) is used to calculate stress changes associated with construction or heating. Second, shear and normal stress criteria for creation of new fractures and/or opening of pre-existing fractures are applied, and permeability changes of individual fractures or sets of fractures are estimated. A literature review shows that permeabilities are sensitive to changes in shear and normal stress, but little direct experimental data quantify the effect of stress changes or heating on permeability changes. Predictions of permeability are therefore based indirectly on the effects of stress on fracture aperture, and a cubic law relation between the aperture and transmissivity. Note that permeability of fractured rock masses is often dominated by preferential flow paths. Third, a network flow model (FracMan, MAFIC) is applied to estimate the change in permeability of the rock mass. This procedure is recommended because the comprehensive literature review shows it to be consistent with availability of laboratory and field data and numerical models.

The choice of the proposed continuum modeling codes FLAC-3D and ABAQUS builds on results from the DECOVALEX program, which evaluated the performance of these codes in simulating experimental and field data, which include coupled thermal-mechanical-hydrological processes.

The input data required by the stress modeling codes include the in situ stress, mechanical and thermal properties of tuff, mechanical properties of fractures, and fracture statistics. These data are available for the repository site. Available information about permeability can also be incorporated.

Activity 1.10.4.3.2 - Borehole damage analysis. The objective of this activity is to provide information on the potential for static loads on the waste package and for radionuclide releases caused by spalling or breakup of the wall of the emplacement drift or borehole.

No progress was made during this reporting period; this was an unfunded activity.

Activity 1.10.4.3.3 - Geomechanical properties analysis. The purpose of thi[illegible] vity is to perform experimental investigations of the coupled thermal-mechanical, thermal-hydrological, and thermal-chemical response of the rock to conditions similar to the near-field environment of the proposed nuclear waste repository. This activity includes testing of 0.5-m-scale blocks of tuff. Data at this scale are needed to provide input to models used for analysis of the repository because very few data sets are [illegible]lable from in situ rock masses and data from smaller samples commonly tested in the labora[illegible] do not provide information on fracture behavior. Moreover, in tests at this scale known bounc[illegible]ry and environmental conditions can be impo[illegible]ed on a rock sample that contains multiple fractures; field data are often poorly constrained because of inherent limitations on boundary conditions, sampling intervals, and material characterization.

Rock Mass Testing

An experiment was conducted on a 0.5-m scale block of Topopah Spring tuff that contained an artificial, horizontal fracture. The experiment was conducted at effectively zero fracture interface stress and at room temperature. The purpose of this test was to establish a standard of reference for subsequent experiments at elevated stresses and temperatures. The sample was prepared using two right prism blocks of Topopah Spring tuff having typical edge dimensions of 25 cm. Fluid flow was generated by a point source in the plane of the fracture at its center, connected to a pressurized fluid reservoir using a small diameter tubing. This configuration creates a radial flow field to allow probing the effect of anisotropy of the rock fabric on the flow in the fracture. Fluid flow was monitored at 38 locations at intervals of about 2.5 cm along the perimeter of the fracture.

Results of the experiment can be summarized as follows:

- Imbibition was the primary fluid sink mechanism early in the experiment. This result was expected because capillary pressures in the material typically are significantly larger than one atmosphere.

- The fracture surface in the sample had no natural hydrologic fast paths.

- Essentially all the measured flow at zero stress for this sample was through the fracture and into 8 of the 18 channels. Three-fourths of the channels reported no flow for the nominally isotropic fracture aperture. No flow through the faces of the blocks away from the fracture plane was observed in this short experiment. This is consistent with the analyses of pore structure in this material.

- Recorded flow was dominated by paths generally parallel with the observed anisotropy of the rock fabric.

The results indicated that the impedance of a flow path may change with time and, as expected, relatively low driving pressures are sufficient to cause flow through this smooth fracture at zero axial stress.

The method of collecting fluid at many discrete locations along the fracture boundary appears to be a promising method of quantifying fluid flow through a fracture. In particular, quasi-quantitative measurements likely can be made of the effect of the anisotropic rock fabric on the flow as a function of compressive stress and temperature.

Support of the Single-Heater Test in the ESF

A reflective optical extensometer instrument with improved performance was installed in the single-heater test that is being conducted in the ESF. The reflective optical extensometer system is being used to monitor deformation in the horizontal plane and in the direction perpendicular to the heater. The system was used to monitor position of reflective anchors placed at distances of approximately 2.5 m and 4.3 m from the heater, for several months starting in late August 1996 and continuing through January 1997. The system has operated nearly continuously over several months in a location that was nearby to continuous mining operations including drill and blast, and mechanical excavation activities. Thus, the system has been found to be rugged enough to hold up over extended periods in the underground environment

While the reflective optical extensometer system operated successfully over this extended period, the quality of the data is not as good as expected. Data collected using the system show extension over the measured interval, which is consistent with analytical predictions of the test, and with data observed using conventional multipoint borehole extensometer systems in boreholes parallel with the reflective optical extensometer hole. While data from the reflective optical extensometer system are within an order of magnitude of the values observed by the conventional system, a higher precision result was expected, but was not obtained. This result is disappointing because the instrument is capable of precision in the range of 50 microns. This problem is currently being investigated.

In addition, demonstrated was the ability to make a measurement over a length of approximately 3 m with one anchor in the hole. However, when multiple anchors were placed in the hole, the shallower anchors blocked out some of the return beam from the deep anchors. Even though a detectable beam was returned from the deepest anchors, the signal level for them was below that required by the phase laser instrument. Work is in progress to increase the laser power and to improve the beam collimation to eliminate this problem.

<u>Support of the Large Block Test</u>

Geomechanics work on the large block test included the installation of three separate systems for monitoring deformation on the large block. These included 18 three-component fracture monitors mounted across significant fractures on the surface of the block. Conventional 4-anchor multipoint borehole extensometers were installed in 6 boreholes, and reflective optical extensometers were installed in 2 holes. All instrumentation is currently operational and collecting data

Initial deformation data from the multipoint borehole extensometer and fracture monitor instrumentation show that expansion started soon after the start of heating. The multipoint borehole extensometer anchors are spaced evenly in the hole, and these data show that most of the deformation is occurring between the second and third anchors, which corresponds to a vertical plane bisecting the block. Comparison with fracture maps indicates that a fracture zone is located in this region, and the preliminary conclusion is that the heating is causing fractures to open in this unconfined environment.

Forecast: Work on block stability analysis will continue to focus on evaluating available methods for estimating changes in fracture permeability surrounding drifts in the ESF and repository in response to construction-induced stress changes and, subsequently, in response to the thermal pulse arising from waste emplacement. These results are needed for modeling changes in repository-level moisture movement Selected techniques will also be used to estimate bounds on the changes in the permeability of the host rock resulting from drift excavation and from thermomechanical stresses. Experimental data and data from the ESF will be used as they become available. To the extent possible, correlations will be made with matrix rock properties, fracture patterns, or other parameters readily measured in the ESF or the laboratory and available from other activities.

Block stability analysis work will also include analysis of deformation data from the single-heater and large block tests and simulation of the drift-scale test in both two and three dimensions to assess the performance of the thermal-mechanical models.

Support of the single-heater test will include continued monitoring of rock deformation using the current reflective optical extensometer instrument.

Support of the large block test will include continued monitoring of rock deformation using the reflective optical extensometer system, conventional multipoint borehole extensometer instrumentation, and three component fracture monitors installed on the surface of the block.

Rock mass testing will include performing laboratory experiments on 0.5-m-scale blocks of Topopah Spring Tuff to obtain data on coupled thermal-hydrological mechanical-chemical processes in a rock specimen containing multiple fractures These experiments will provide data on the effect of temperature and stress on fracture permeability and on the coupled processes occurring in the fractured rock at temperatures in the 90 to 250°C range. These tests coupled with the single-heater test now under way in the ESF will provide the only experimental data (before the total system performance assessment that will support the viability assessment) on nonisothermal flow at a scale large enough to include multiple fractures. These 0.5-m-scale laboratory tests also allow the investigation of the effects of thermal and mechanical stresses on flow, providing a degree of test control that is not achievable in situ. The results will be essential for constraining models of near-field coupled processes that support total system performance assessment and waste package design. Moreover, data from these experiments will help determine pre- and postclosure rock characteristics for models of pre- and postclosure repository performance. Data will also be used in the design and interpretation of the drift-scale test now being constructed.

5.2.5 Study 1.10.4.4 - Engineered Barrier System Field Tests

The laboratory tests described in Studies 1.10.4.1 through 1.10.4.3 (Sections 5.2.2 through 5.2.4 of this progress report) require validation by in situ field tests in the repository horizon to establish the applicability of the laboratory studies to the repository block. The objective of this study is to investigate the geomechanical and geochemical behavior and movement of water in the rock mass under the influence of the thermal loading of the waste package. The study will investigate heat-flow mechanisms, fracture aperture change, geochemical reactions, the relationship between boiling and dryout, and the rewetting of the dryout region when the repository is cooled down. Coupling between heat, hydrology, geomechanics, and geochemistry will be included in the study. These activities will test some of the coupled processes that will be part of the models the program plans to use to predict the repository near-field environment.

This study had three activities in the SCP. When the study plan was written and subsequently revised, the material was organized differently than in the SCP. The cross linkage of the original activity descriptions to the current activities is included in Appendix A of this progress report. The status of the current activities is provided below.

Some of the material that follows discusses field tests. Other parts discuss laboratory tests in support of the field testing. In addition, some of the laboratory test results provide important information to support analysis of near-field rock properties. In turn, these analyses will be used in repository design and performance assessment.

Activity 1.10.4.4.1. Sampling and sample analyses. The objective of this activity is to collect and analyze material samples (rock, gas, and water) before, during, and after heating of the rock as field testing occurs. The laboratory analyses of the samples will determine hydrologic and geochemical properties of the rock and chemistry of the gas and water.

Cores resulting from the installation of the single-heater test have been obtained for mineralogical and petrological analyses. The analyses continued this reporting period, and the results will be reported by the end of FY 1997.

Activity 1.10.4.4.2 - In situ testing. The objectives of this activity are to develop detailed planning documents for the engineered barrier system field tests and analysis, to check out and debug techniques and hardware, to perform comparative evaluations of candidate test component methods, to procure equipment, to purchase or manufacture test components, to calibrate and install test components, and to conduct in situ testing.

Exploratory Studies Facility Thermal Tests

Construction of the single-heater test began in August of 1995, and the heater was energized on August 26, 1996. Pre-heat data were collected several days before the heater was activated. The boiling point isotherm around the heater is at 0.7 to 0.8 m radial distance. The data collected from the single heater test has been reported in the interim report (CRWMS M&O, 1997u).

The preliminary results of the coupled thermal-hydrological-mechanical-chemical responses of the heated rockmass indicate that the heat moved the moisture away from the heater borehole. As of January 30, 1997, a small dryout region may have been created around the heater. The primary purpose of the single-heater test is to test thermal-mechanical responses of the rock mass. Therefore, to avoid interference with the thermal-mechanical holes, the boreholes for the coupled thermal-hydrological-mechanical-chemical processes were not located near the heater borehole. Thus, the small dryout region is not well monitored, and its existence will need to be verified later, assuming it expands.

Also the water that has been relocated by the heat is more diluted than the local ground water and may have only reached chemical equilibrium with the secondary minerals on the fracture surfaces. As a result, the chemistry of this relocated water is likely to be substantially different from the chemistry of J-13 water. The thermal-mechanical measurement results are not conclusive enough for assessing thermal-mechanical-hydrological couplings. A complete analysis of the data will be conducted when the heating phase of the test is completed.

Construction of the drift-scale test continued. Instrument installation has begun. The RTD and Teflon™ tubes, as neutron logging hole liners, were installed on February 28, 1997, in the ESF-HD-TEMP1, which is the north longitudinal hole parallel to the heater drift. Multipoint borehole extensometer anchors with Teflon™ liners to seal the hole have been installed in the nearby, parallel ESF-HD-MPBX1 hole.

Large Block Test

The large block test instrument installation was completed in February 1997. The instruments included the RTD to measure temperature both in boreholes and on the block surface, multipoint borehole extensometer and optical multipoint borehole extensometer, fracture gauges on the block surface, humicaps to measure relative humidity, pressure transducers to measure gas pressure, ERT electrodes, Teflon™ liners in neutron holes, and Pyrex™ liners in the observation holes. Coupons of carbon steel and introduced material in the ESF were also installed in the packers along with the humicaps and pressure transducers. The large block surface is covered with a layer of moisture barrier, and three layers of insulation materials: one layer of ultratemp, one layer of R-19 building insulation, and one layer of reflective insulation.

The ambient data acquisition began the week of February 17, 1997. The heaters in the large block test were energized on February 28, 1997.

ʳacture Flow Versus Matrix Imbibition

A test was conducted to investigate the effect of water head on the fracture flow through a thermal gradient. In the previous test described in Progress Report 15, the water head was 1.45 meters above the top of the fracture, and no water was observed to pass through the heated zone. In the test this period a water head was generated at 2.92 m above the fracture. Although neither the sample nor the thermal conditions in the sample were changed, the water was able to flow through the boiling zone in less than 0.7 hours in the second test conducted with the higher head. There was little imbibition of water into the matrix during the flow in the fracture. This test result indicates that the infiltration rate has a strong effect on the fracture flow.

Matrix Permeability

Gas permeability measurements on the N-1 core sample were completed. Resulting data will be analyzed to determine the Knudsen diffusion coefficient.

Activity 1.10.4.4.3 - Pre- and post-test calculations. The objective of this activity is to perform scoping calculations in support of engineered barrier system field test design, planning document development, and reducing and analyzing test data. This activity includes the verification and validation process necessary to qualify the numerical methods to be used if not already accomplished by another activity.

No progress was made during this reporting period; this was an unfunded activity.

Forecast: The heating of the large block test will continue throughout most of the next reporting period. Once erior block temperature has reached approximately 140°C, with the temperature at the op of the block kept near 60°C, these conditions will be held stable for about a month. After that time, the heaters will be turned off to start a cool-down phase. Data acquisition will continue during the heating and cool-down phases.

The heating phase of the single-heater test will continue until about the end of May 1997. A decision then will be made whether to continue the heating phase for another three months or to turn the heater off to start a cool-down phase. If a cool-down phase is started at the end of May 1997, it may last throughout the next reporting period. Data acquisition will continue during both the heating and cool-down phases.

Installation activities at the drift-scale test will continue. By the end of this fiscal year most of the instruments should have been installed in boreholes in the drift scale test.

The laboratory characterization of the hydrological properties of the core samples from the drift-scale test will continue.

The fracture flow versus matrix imbibition experiments in the laboratory will continue with focus on the effect of fracture aperture and water head on the fracture flow and matrix imbibition.

The laboratory experiments on fracture healing will continue. The focus of this work will be the effect on fracture healing of the volume of water flowing through a sample.

The design of an experiment to investigate enhanced vapor diffusion in Topopah Spring Tuff will be completed. The goal of this experiment is to understand the process of vapor diffusion in tuffs at Yucca Mountain, its role in the movement of heat and moisture, and conditions under which vapor diffusion might be enhanced.

5.2.6 Study 1.10.4.5 - Characterize the Effects of Introduced Materials on Water Chemistry in the Postemplacement Environment

The objective of this study has been to identify significant chemical modifications of the near-field environment to the chemistry that would be expected under thermally perturbed geological conditions. The modifications are caused by the construction and operation of the repository. The geological conditions are defined by Study 1.10.4.1, discussed in Section 5.2.2 of this progress report, but nonredundant studies are being conducted under the waste package technical area. A complete picture of the modified chemical and hydrological system includes, in addition to construction materials, introduced air and water, crushed tuff or muck rock used as backfill or invert material, and introduced or enhanced microbial populations.

This study did not exist in the SCP. It was developed from aspects of the near-field geochemistry study (Section 5.2.2 of this progress report) which included the effects of introduced materials and radiation from the waste packages. The original draft study plan included four activities, but when the draft study plan was revised, the material was organized differently than in the first draft. To avoid confusion between similarly numbered activities in the initial draft [which was captured in the Site Design and Test Requirements Document (DOE, 1995f)] and the next draft, the current activities were renumbered with higher numbers. The cross linkage of the original activities to the current activities is included in Appendix A of this progress report. The status of the current activities is provided below.

Activity 1.10.4.5.5 - Integration: Program Planning; identification, characterization and screening of materials; and bibliographic maintenance and literature review. The objectives of this activity are to prepare planning documents for the introduced materials study, to develop a list of materials that might be used in the repository (including locations, quantities, and concentrations), to develop a chemical data base regarding the materials, to rank the materials on the basis of aggressiveness under expected and certain unexpected repository conditions, to identify materials for which information is inadequate, and to gather, synthesize, and evaluate data from the literature. These objectives are not necessarily sequential, and some products will be updated throughout the study.

This study is integrated closely with Waste Package study "Thermal and Chemical Degradation of Concrete and Invert Material," reported in Section 6.?.1 of this progress report to prevent duplication of effort. This work has also supplied information to the Near-Field Environment Performance Assessment Workshop, and to the Near-Field and Altered-Zone Environment Report (Wilder, 1996). Work on this activity during this reporting period consisted of support for the development of abstraction and testing methodologies for the total system performance assessment.

Activity 1.10.4.5.6 - Solubility and stability experimental studies at ambient and elevated temperatures. The objective of this activity is to conduct dissolution and precipitation kinetics experiments to determine the sensitivity of the kinetics to temperature and fluid composition. Stoichiometric and nonstoichiometric dissolution of introduced materials, in saturated and unsaturated environments, will be addressed. The experiments are intended to identify the dissolution precipitation mechanisms, the effects of solid solution on rates of dissolution and precipitation, the solid reaction products, and the resulting water chemistry. Solid, liquid, and gas phase stability will be addressed.

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.5.7 - Chemical reactivity stability experimental studies at ambient and elevated temperatures. The objective of this activity is to conduct chemical reactivity experiments on soluble products of introduced solid phases, on introduced organic and inorganic fluids, on introduced material interactions with water and vapor in the presence of a radiation field, on the potential effects of introduced materials on predicted natural chemical reactions, and on the significance of natural mineral moderation (e.g., zeolites and buffering effects) on the aggressiveness of introduced materials.

The construction material of specific interest this reporting period was concrete because of its potential as a structural support in the emplacement drifts. This set of experiments was intended to provide data for a quick engineering assessment of the microstructural, mineralogical, and (to a lesser extent) mechanical changes in concrete and changes in associated water chemistry resulting from a repository hydrothermal cycle. The concrete samples that have been used in these experiments were intended to support a design decision regarding the use of precast concrete liners for mechanical support in repository emplacement drifts. In such a location, the concrete would be subjected to elevated temperatures of at least 150 to 200°C and perhaps even greater temperature if backfill is used. The objectives of the work have been to conduct hydrothermal alteration tests in the laboratory, emplace samples in field tests (large block test, single-heater test and drift scale test) and monitor the use of construction materials in both types of tests. Concrete coupons, both alone and sandwiched with potential waste package materials, have been installed in the single-heater test. Because of the smaller chamber size of the large block test packers, only sandwich coupons were emplaced in that test. This work also includes observation during the operation of the heater, and collection and follow-up studies after the test.

In addition to the work just described, concrete invert from the ESF (fibercrete from the north portal) has been used in a study of vapor and aqueous phase alteration at 200°C. Three experimental runs of progressively longer duration have been initiated. The first batch of samples (one month duration) has been removed and the analyses (mineralogical, chemical and mechanical) are complete. A milestone report on the results for the first batch of samples (Meike, 1997a) has been submitted. In this relatively short-term experiment, evidence of alteration, sometimes extensive, was found, especially for the vapor phase samples. Some samples contain crystalline Ca-Si-hydrates. The analysis of the untreated samples clearly shows that carbonates make up a large proportion of the shotcrete and invert aggregate, even the sand sized particles. This would significantly alter the results of chemical modeling studies that have used a 100 percent quartz sand composition [e.g., the backfill study by Meike and Glassley (1997)].

The results of analyses of the first batch of hydrothermally treated samples demonstrate that alteration occurs even in as little as a month at 200 to 251°C, and in some instances the alteration is extensive. Further interpretation of this data would be premature. Detailed interpretation of these results will be made after the results of longer-duration treatments have been analyzed and trends can be established.

Activity 1.10.4.5.8 - Colloid stability experimental studies at ambient and elevated temperatures. The objective of this activity is to identify introduced materials that can produce colloids, and examine the nature and stability of the colloids. This activity is intended to complement other work being conducted in Study 8.3.1.3.5.2 - Section 3.2.11 of this progress report.

No progress was made during this reporting period: this was an unfunded activity.

Activity 1.10.4.5.9 - Biodegradation stability experimental studies at ambient and elevated temperatures. The objective of this activity is to identify and characterize microbes that might be introduced into the repository, and microbes (both native and introduced) that derive nourishment from introduced materials that could be brought into the repository. The activity will identify introduced materials that will encourage microbe growth, identify chemical products of microbial degradation, and identify and evaluate the potential for introduction and growth of microbes from external sources. This activity is intended to complement other work being conducted in Study 8.3.1.3.4.2 (Section 3.2.9 of this progress report).

With respect to microbe studies, the objectives have been to use the large block test and the drift-scale test for migration and survival studies supported by appropriate laboratory experiments. In this work, microbes were obtained from the large block test site and labeled by a method suitable to the application (Meike, 1996; Meike, 1997b; Meike and Horn, 1997). Tests were conducted to understand the longevity and thermal stability of the methods. Suitable installation tools were designed and manufactured. Care was taken that the methods and practices did not interfere with other tests.

As part of this work, drug-resistant and fluorescent-dye-labeled bacteria have been developed to study microbial survival and migration, and preliminary longevity and thermal stability tests have been conducted. Survival specimens have been placed at intervals along three vertical boreholes in the large-block test to take advantage of the thermal gradient that will be present after the heater is powered. And, because of variations in placement and coupon design, the bacterial samples in each hole are expected to experience different, but relevant chemical environments. The migration experiments were initiated by extruding a microbe-inoculated gelatinous medium into the heater holes before the insertion of the heater.

These studies will also be performed in conjunction with the drift-scale test.

Activity 1.10.4.5.10 - Historical analogs. The objective of this activity is to identify sites of interest as historical analogs to Yucca Mountain (determined from the materials list developed in Activity 1.10.4.5.5); to collect samples from these sites; to analyze the samples for the information identified in Activity 1.10.4.5.5; to provide constraints for the experiments in Activities 1.10.4.5.7, 1.10.4.5.8, and 1.10.4.5.9; and to provide long-term data not obtainable from experiments for the development of the introduced material-rock-water interaction simulation activity (1.10.4.5.11).

No progress was made during the reporting period; this was an unfunded activity.

Activity 1.10.4.5.11 - Computer modeling and code development. The objective of this activity is to develop the necessary codes (if not otherwise available) and to conduct predictions and simulations of experiments, natural analogs, and repository performance with respect to introduced materials effects on the near-field environment. Validation of developed models is included in this activity, which is complementary to Study 8.3.4.2.4.1.

No progress was made during this reporting period; this was an unfunded activity.

Forecast: Hydrothermal experiments on the alterations of concrete will continue for the longer duration tests. These samples will be removed and analyzed at the appropriate time. A more pronounced alteration is expected in the longer term samples than those that have been analyzed to date.

Concrete coupons that have been installed in the large block test and the single-heater test will be removed and analyzed after the cool-down period. Water that has been found in a single-heater test neutron-logging hole will be collected and analyzed.

Data and modeling work will be compiled on concrete-water systems that has been laid aside after the termination of the International Program's Fundamental Materials Task.

Work has just been initiated on the collection of microbes for the drift-scale test. These microbes will be isolated, labeled, and installed for later removal and analysis at the termination of the test.

5.2.7 Related International Postemplacement Near-Field Environment Work

No progress occurred during the reporting period. As of November 8, 1995, the subsidiary agreements under which the cooperative work had been conducted were terminated, and all international collaboration was discontinued.

The Office of Civilian Radioactive Waste Management has bilateral agreements with Canada (Atomic Energy of Canada Limited [AECL]), Switzerland (S[illegible] National Cooperative for the Storage of Radioactive Waste [NAGRA], and Sweden (Swedi[illegible]clear Fuel and Waste Management Company [SKB]) and has participated in activities of int[illegible]nal organizations including the Organization for Economic Cooperation and Developm[illegible]ar Energy Agency (OECD/NEA), the European Commission (EC), and the International Atomic Energy Agency (IAEA).

Forecast: No international work is presently planned.

5.3 CHARACTERISTICS AND CONFIGURATIONS OF THE WASTE PACKAGES PRECLOSURE (SCP SECTION 8.3.4.3)

The purposes of this study are as follows:

1. To ensure that the waste package design complies with preclosure design criteria of 10 CFR 60.135 through analysis of the design and comparison of the design with the regulatory criteria

2. To obtain information on the physical characteristics of spent nuclear fuel and high-level waste that is to be accepted into the Civilian Radioactive Waste Management System (CRWMS). This information will include information on the suitability of a given waste form for emplacement, receipt rate projections, etc.

5.3.1 Information Need 2.6.1 - Design Information Needed to Comply with the Preclosure Criteria from 10 CFR 60.135(b)

The purpose of this activity is to obtain information necessary to ensure and verify that the waste package design complies with the specific design requirements of 10 CFR 60.135(b)(1) through (4). These requirements restrict the presence of explosive, pyrophoric, and chemically reactive materials in the waste packages. They also restrict the presence of free liquids in the waste package. Finally, they include requirements regarding the ability of the waste package to withstand handling loads and for waste package labeling.

Progress Report #13 (DOE, 1996f) discussed the compliance of light water reactor fuel and radioactive waste glass with the requirements of 10 CFR 60.135(b)(1) and (2). Chemical stability of several types of DOE-owned spent nuclear fuel is being investigated outside the Project. The fuel types include Fort St. Vrain fuel and three types of Shippingport fuel. The graphite blocks and graphite and silicon carbide coatings of the Fort St. Vrain fuel are not chemically reactive.

To ensure that the DOE-owned spent nuclear fuel and the waste package designs comply with the specific requirements of 10 CFR 60.135 (b), Project staff met this reporting period with representatives of the DOE National Spent Fuel Program, the Office of Naval Reactors, the Idaho National Engineering and Environmental Laboratory, the Hanford Site, and the Savannah River Site to identify and discuss the information needed to demonstrate compliance with the requirements of 10 CFR 60.135(b) regarding explosive, pyrophoric, and chemically reactive materials; free liquids; handling; and unique identification. Because of the wide variety of fuel types, operating histories, and storage conditions, it will take a considerable period of time to compile the required information and/or perform the necessary characterization tests. Plans and schedules for acquiring the information are being developed and integrated into the budgeting process.

The materials referenced in Section 5.1.4 of this progress report are standard engineering materials. This may include helium, an inert gas, as a fill gas. None of these materials is explosive, pyrophoric, or chemically reactive.

No work was scheduled on methods to uniquely identify the waste packages to address 10 CFR 60.135(b)(4); that was an out-year activity. Because this requirement does not affect the viability assessment, no work is planned to address it before FY 1999.

<u>Forecast</u>: Work on the presence of explosive, pyrophoric, and chemically reactive materials and on free liquids is complete for light water reactor fuel, high-level waste glass, and disposal container materials. Additional analyses of DOE-owned spent fuel will be performed in FY 1997. Additional structural analyses for design basis rockfalls have been delayed because rockfall data are still being acquired, so the rockfall size and frequency cannot be specified at this time. The Project is currently considering when to complete an analysis of the latest rockfall data that will provide size and frequency distribution.

5.3.2 Information Need 2.6.2 - Design Information Needed to Comply with Preclosure Criteria from 10 CFR 60.135(c) for Waste Forms

The purpose of this activity is to obtain information necessary to ensure that high-level waste forms to be emplaced at the repository comply with the specific requirements of 10 CFR 60.135(c). This regulation requires that the waste forms be in solid form in sealed containers and that particulate waste forms be consolidated. It also limits the presence of combustible wastes.

Progress Report #13 (DOE, 1996f) discussed the compliance of light water reactor fuel and high-level waste glass with the requirements of 10 CFR 60.135(c). Studies of the chemical reactivity of certain types of DOE-owned spent fuel are discussed in Section 5.3.1 of this progress report.

Forecast: Work on this activity is complete with regard to light water reactor fuel and defense high-level waste glass. Additional analyses of DOE-owned spent fuel characteristics will occur in FY 1997.

5.3.3 Information Need 2.6.3 - Waste Suitability

The objective of this activity is to develop the specifications for determining the suitability of spent nuclear fuel and high-level waste forms for emplacement at a repository.

On the basis of the work reported in Section 5.3.2 of this progress report, light water reactor fuel and defense high-level waste glass were determined to be generally suitable for disposal with regard to compliance with 10 CFR 60.135(c).

Information on the characteristics of the waste continues to become available as additional fuel is discharged and the system architecture is defined and refined. Historical and projected spent nuclear fuel discharges are updated annually by the DOE Energy Information Administration. These data are used by DOE in a variety of ways. Because these data contain not only discharge quantities but also spent nuclear fuel characteristics, the data bases form the foundation of any waste stream analysis.

System models are used to simulate the schedule and rate of pickup from utilities and to predict how the fuel will be containerized using container technology assumptions (capacity and heat limits) and the fuel characteristics. The system models also simulate movement of every container and spent nuclear fuel assembly through the CRWMS, resulting in a repository arrival profile. Since the last progress report, a Waste Quantity, Mix, and Throughput Study (CRWMS M&O, 1997j) has been conducted that generated updated logistics arrival profiles that will be used as input to the design to support the viability assessment. The arrival profiles were generated for commercial spent nuclear fuel, defense high-level waste, and DOE-owned spent nuclear fuel based on reasonably bounding assumptions regarding waste amounts, waste pickup schedules, and transport scenarios.

An activity was initiated to define waste acceptance criteria for material to be accepted by the MGDS for disposal. The Waste Acceptance Criteria Document, Revision 0, will define preliminary acceptance criteria for commercial spent nuclear fuel, high-level radioactive waste, and canisters/casks with a focus on material, physical, dimensional, chemical, thermal, and radiation criteria. The purpose of Revision 0 (to be completed next reporting period) is to provide a comparative tool largely based upon current repository capabilities as identified in the reference design by which outlier DOE-owned or commercial fuels can be identified for focused attention prior to the viability assessment. The Waste Acceptance Criteria Document will evolve over time to capture the negotiated interface criteria adjustments to the design or waste forms to facilitate waste acceptance.

Forecast: The DOE is currently pursuing a strategy that will use private regional servicing agents to provide waste acceptance and transportation of commercial spent nuclear fuel. Future work on waste acceptance will be contingent upon further definition of tasks to support the regional servicing agents concept.

An annotated outline for the Waste Acceptance Criteria document will be completed by the end of June 1997. Revision 0 to the document will be completed by the end of September 1997.

5.4 WASTE PACKAGE PRODUCTION TECHNOLOGIES (SCP SECTION 8.3.4.4)

5.4.1 Design Activity 4.3.1.1 - Waste Package Fabrication Process Development

The objective of this activity is to determine the processes to be used in fabricating the components of the waste packages other than the waste form itself. The fabrication techniques selected will be important in constructing a waste package that will meet repository containment and waste isolation performance objectives, because degradation mechanisms such as corrosion are often sensitive to fabrication techniques.

Possible processes for material and manufacturing continue to be investigated through discussions with various manufacturers and vendors. This is an ongoing process.

Forecast: The Waste Package Fabrication Process Report (CRWMS M&O, 1996x) will be updated late in FY 1997. This report will be supported by the fabrication methods evaluated by fabrication of full circular mockups The closure weld development will address the manner of forming the thick-walled outer barrier cylinder, including any weld seam designs that may be needed to complete this component. This mockup will be used to evaluate the shrink fit design including any crevices between the barriers and the thermal properties.

5.4.2 Design Activity 4.3.1.2 - Waste Package Closure Process Development

The objective of this activity is to determine, by using the logical sequence described for this issue, the process to be used in the final closure of the disposal containers. The closure process is vital to waste package performance. Inadequate closure or improper closure techniques could reduce the ability of affected waste packages to contain and isolate waste.

Nevada Line Procedure-7-3, which describes how the closure process development programs will be conducted, was developed and approved. The Waste Package Closure Methods Technical Guidelines Document (CRWMS M&O, 1997o) was written. The actual development work for this program started in March 1997.

Forecast: The closure weld development program will be completed. This work will include the manufacture of a full circular mockup, approximately one third as long, using the shrink fit method of manufacture. The mockup will be tested for thermal conductivity and presence of a crevice between the two barriers. The two lids will be welded into place, and stress measurements will be taken both during the test and at the completion. The report for this program will be completed and submitted.

5.4.3 Design Activity 4.3.1.3 - Waste Package Closure Inspection Process Development

The objective of this activity is to determine, by using the logical sequence described for this issue, the process to be used in inspecting the final closure of the disposal containers to ensure the closure is adequate, in compliance withe design, and supportive of meeting repository overall performance objectives for waste containment and isolation.

This activity will be conducted in parallel with the closure weld activity. The Nondestructive Examination Development Technical Guidelines Document (CRWMS M&O, 1997p) has been written and the development program started in March 1997.

Forecast: The nondestructive examination program will be completed. The effort will include a method and mapping of the crevice between the two barriers and the actual nondestructive testing of the two lid welds. A report detailing this phase of the program will be generated.

5.4.4 Design Activity 4.3.1.4 - Remote In-Service Inspection Development

The purpose of this activity is to develop an effective method and program for remote in-service inspection of the waste package. Remote inspection techniques will be needed to monitor waste package performance and condition during the preclosure period.

No progress was made during this reporting period; this was an out-year activity. Related work on performance confirmation concepts is discussed in Section 6.15 of this progress report.

Forecast: No activity is forecast for in-service inspection program development because of its relatively low priority, given the length of time to beginning of emplacement.

5.4.5 Design Activity 4.3.1.5 - Internal Filler Material Process Development

The purpose of this activity is to determine a method of providing effective insertion of filler material into the waste package. Filler material may be used to absorb neutrons and/or to displace water from the waste packages. Fillers may thereby assist in criticality control.

No further activity is planned for this program. A feasible process for insertion of filler material has been adequately demonstrated into the waste package, should use of filler be determined necessary or appropriate. Disposal containers for canistered fuel may require filler material, but no additional work on filler material is planned because of the Project emphasis on uncanistered fuel.

CHAPTER 6 - PERFORMANCE ASSESSMENT PROGRAM

INTRODUCTION

Performance assessment is the process of quantitatively evaluating component and system behavior, relative to containment and isolation of radioactive waste, to determine compliance with the numerical criteria associated with 10 CFR Part 60 and 10 CFR Part 960. Performance assessment includes evaluations (a) of the preclosure radiological safety of the public and workers and (b) of the postclosure waste isolation performance of the Mined Geologic Disposal System (MGDS). Performance assessment supports evaluations of site suitability, the design of the MGDS, and evaluations of regulatory compliance. A major aspect of performance assessment involves mathematical modeling of natural events, as well as processes and events induced by the construction and operation of the MGDS. The mathematical models are developed and validated and the computer codes that implement the mathematical models are verified, benchmarked, and documented. Laboratory and field measurements and experiments provide the understanding and data needed to develop and validate the mathematical models.

Current and Future Focus

A key element in performance assessment is total system performance assessment, which evaluates the postclosure waste isolation performance of the combined natural and engineered barrier systems for expected and unexpected events and processes to demonstrate compliance of a potential MGDS at Yucca Mountain with the quantitative criteria of 10 CFR Part 60 and 10 CFR Part 960. Total system performance assessments are conducted iteratively to reflect the evolution of site data, the design of the engineered barrier system, the understanding of natural and engineered barrier system processes, and the development of conceptual and mathematical models, including associated computer codes. The latest assessment was Total System Performance Assessment - 1995 (CRWMS M&O, 1995e), which was summarized in Progress Report #14 (DOE, 1996g). In contrast to earlier assessments (Barnard et al., 1992; Eslinger et al., 1993; Andrews et al., 1994; Wilson et al., 1994), this assessment included the effects of long-term periodic climatic changes, more representative conceptual models of the site hydrogeology and more realistic models of engineered system behavior. Current efforts are focused on preparing for the next total system performance assessment, planned for the 1998 time frame, in support of the viability assessment (see Section 1.2.2 of this progress report).

During the reporting period, the major performance assessment emphasis was on (a) additional preliminary preclosure radiological safety analyses of the advanced conceptual MGDS design, (b) the beginning of a waste retrievability study to identify retrievability-related MGDS design requirements, (c) workshops to define efficient and valid abstractions of detailed models of natural and engineered barrier system processes for total system performance assessment, (d) continued waste form and waste container material experiments and related modeling to determine waste release rates and container degradation rates in various potential near-field repository environments, (e) the preparation of a Performance Confirmation Plan to identify performance confirmation activities after the submittal of a license application to the U.S. Nuclear Regulatory Commission (NRC), and (f) continued evaluations of the potential impacts of site characterization activities, including the construction and operation of the

Exploratory Studies Facility (ESF), on the postclosure waste isolation performance of a potential repository.

Following is a summary of the major performance assessment activities and results of the reporting period.

Preclosure Performance Assessment

The meteorological program has collected and analyzed additional wind data to allow more accurate predictions of dispersion and extreme conditions than previously possible. Evaluation began of commercially available software to allow local topography to be considered in dispersion calculations.

A draft outline of the Preclosure Radiological Safety Chapter (Chapter 7) of the Project Integrated Safety Assessment was produced.

Waste Retrievability

A waste retrievability study is being conducted to develop the technical rationale for the MGDS design approach to be used for complying with the 10 CFR Part 60 requirements related to retrievability. This study will also identify potential scenarios concerning the final disposition of the retrieved waste. The Retrievability Strategy Report (CRWMS M&O, in prep.[g]) will be completed in April 1997, and the related MGDS retrieval design activity will be completed at the end of fiscal year (FY) 1997.

Higher-Level Findings and NRC Siting Criteria

There was no activity with respect to formulating higher-level findings in accordance with 10 CFR Part 960 and with respect to the NRC Siting Criteria of 10 CFR 60.122. Refer to Section 2.2.1 of this progress report for relevant regulatory activities.

Total System Performance Assessment (including Ground-Water Travel Time, Individual Protection, and Ground-Water Protection)

Total system performance assessment activities concentrated on preparing, conducting, and evaluating the results of several workshops to facilitate the abstraction of detailed natural and engineered barrier system process models for inclusion in a total system performance assessment. An abstraction is defined as a simplified/idealized model that reproduces or bounds the essential performance assessment elements of a more detailed process model. The complexity of the repository system, the long regulatory time periods, and the stochastic nature of the standards make the use of three-dimensional process models numerically intractable. Therefore, abstraction of the most sensitive aspects of the problem is a critical element of total system performance assessment model development. The abstraction, however, must capture uncertainty and variability. The abstractions must also be tested against process models to ensure their validity.

During this reporting period, workshops were held on the following seven topics: (1) saturated zone fluid flow, (2) waste package degradation, (3) thermohydrology, (4) unsaturated zone radionuclide transport, (5) waste form degradation and mobilization, (6) near-field environment, and (7) nuclear criticality. Two more workshops are planned for the next reporting period: (1) saturated zone ground-water flow and radionuclide transport and (2) biosphere (which will include radiation doses to humans). The workshops identify (a) issues that need to be addressed by process-level modeling, (b) the detailed processes to be modeled, (c) the abstractions of specific processes needed and the details to be included, and (d) plans for implementing the model abstractions.

Engineered Barrier Materials Experiments and Modeling

Engineered barrier materials experiments and modeling included (a) testing and performance evaluations of spent nuclear fuel and candidate waste container materials and (b) participation in the waste package degradation workshop for developing model abstractions for total system postclosure performance assessments. Summaries of this work follow. Refer to Chapter 5 of this progress report for related work.

Revision 1 of Volume 3 of the Engineered Materials Characterization Report (McCright, in prep.) was submitted to the Yucca Mountain Site Characterization Office (YMSCO) for review and approval. Volume 3 contains the results of testing and modeling activities that have occurred since the report was originally issued in December 1994 as Revision 0 (Van Konynenburg and McCright, 1995). Volume 1 (on the background and history of the engineered barrier system candidate materials) and Volume 2 (on the physical and mechanical properties of the candidate materials) were not revised.

Specimens of Alloy 625 (ASTM B 443), a nickel-chromium-molybdenum alloy, were purchased and added to the corrosion testing program (CRWMS M&O, 1996y), specifically the long-term comprehensive corrosion test, the electrochemically based corrosion tests, microbiologically influenced corrosion tests, galvanic corrosion tests, and the humidity chamber oxidation and corrosion tests.

A new study began in this reporting period to identify effects of the interaction between engineered barrier materials and water or water vapor in the repository. These materials include, in addition to construction materials, introduced air and water, crushed tuff or muck rock as backfill or invert material, and introduced or enhanced microbial populations. In particular, the interest is in those effects that may be outside the bounds of predictions that are based on thermally perturbed rock. The present experiments are intended to support a design decision regarding the use of precast concrete liners for mechanical support in repository emplacement drifts.

Efforts began to both produce and evaluate ceramic coatings for carbon steel applied by various thermal spray techniques. The evaluation includes alumina, titania, combinations of these two materials, and magnesium aluminate spinel. For initial work, aluminum oxide was sprayed using a direct-current electric arc plasma. Die penetrant and metallographic studies began to characterize the resultant coatings and provide a working knowledge of the properties produced under various conditions. Thermal studies were initiated at 300, 600 and 900°C to extrapolate whether phase transformations take place over time at the lower repository temperatures. The relative proportion of phases was determined using x-ray diffraction. After six weeks, no clear indication was found that any transformation had taken place. An experimental matrix was designed for the impact tests on coatings, which will use a 2-m drop tower to simulate rockfall in the repository. A ceramic impactor of appropriate chemistry and density will be used to represent the Yucca Mountain welded tuff. In preparation for corrosion studies to follow, substrates were prepared consisting of cylinders 25 mm (1 in.) in diameter by 150 mm (6 in.) long with hemispherical ends. Suitable racks have been ordered to include these coupons in long-term corrosion studies.

A degradation mode survey on the effects of fabrication and welding on the performance of candidate corrosion-resistant nickel-base and titanium-base alloys was issued (Roy et al., 1996a). This report addresses nickel-base alloys (specifically Alloys 825, G-3, C-4, and C-22) and candidate titanium-base alloys (Grades 12 and 16). Specifically, it provides information on the response of the alloys to the combined effects of temperature, oxidizing or reducing gaseous environments, and pressures inherent in container fabrication and welding operations on the metallurgical phase stability, and mechanical properties. The report also examines the response of the alloys to the expected environmental conditions in the potential repository. One of the important concerns with these high-performance materials is the possible formation of brittle phases, especially in and around the welded regions.

Long-term corrosion testing began last period in eight of the first twelve test vessels. The corrosion-allowance materials, carbon and low alloy steels, were emplaced in the first four test vessels, which contained dilute and concentrated aqueous solutions of near neutral pH at 60 and 90°C. The intermediate corrosion-resistant materials, 70/30 copper nickel and Monel 400, were emplaced in the next two vessels, which contained concentrated acidic solutions (pH 2.6) at 60 and 90°C. The corrosion-resistant materials, the nickel-chromium-molybdenum ~~and~~ titanium alloys, were emplaced in the next two vessels, which also contained concentrated acidic solutions (pH 2.6) at 60 and 90°C. The corrosion-resistant materials will be emplaced in the remaining four vessels, which will contain dilute and concentrated aqueous solutions of near neutral pH. Alloy 625 (ASTM B 443) test specimens were added to the corrosion-resistant materials testing.

Humid air corrosion on salt covered (NaCl) carbon steel (A516 Gr55) specimens was investigated to understand the mechanistic aspects of the degradation process. For testing lasting about 14 days at relative humidities greater than 70 percent and at a temperature of 80°C, salt-covered specimens corrode very fast initially. With time, however, the salt is "consumed" by the oxidation process and the corrosion rate eventually ceases. At longer times the oxide transforms to a more stable oxide and spalls off the vertical surfaces. X-ray diffraction studies were used to try to explain this process. Long-term testing under constant conditions, 80°C and 50 percent relative humidity has begun in an environmental chamber. Specimens include weight loss coupons that are clean, salt-covered, and sandwiched (metal to metal) in order to create crevices. Initial materials being tested are carbon steels, Alloy 625 (nickel-chromium-molybdenum alloy), and a dilute titanium alloy, TiGr12. Numerous specimens are being tested to allow periodic removal for kinetic and mechanistic characterization.

Stress-corrosion crack growth tests using fatigue-precracked and wedge-loaded double cantilever beam specimens began in November 1996. Results obtained so far indicate that Alloy 825 became susceptible to stress-corrosion cracking after exposure to the test environment for 30 and 60 days. Specimens were tested in acidified 5 percent NaCl solutions (pH 2.7) maintained at 90°C. The initial stress intensity was high and ranged from 33 to 52 ksi·in$^{1/2}$. The stress intensity generally decreases as the crack grows. This combination of test conditions is severe, but the intent of this first series of experiments was to discern significant differences in the behavior of the candidate materials. The observed cracking in Alloy 825 appears to follow an intergranular pattern.

Testing of carbon steel specimens in microbially inoculated test cells at room temperature has been completed. The results of these studies were reported in Progress Report #15 (DOE, 1997e); in summary, it was found that a combination of sulfate reducing, iron oxidizing, and slime producing bacteria demonstrated rates of corrosion five times greater than that shown in sterile, abiotic control cells incubated under the same conditions. This same experimental protocol, using the same sets of bacteria, is now being followed to determine corrosion rates of carbon steel test coupons at 50°C.

Electrochemical cyclic potentiodynamic polarization experiments involving iron-nickel-chromium-molybdenum alloys (Alloys 825, G-3, and G-30), nickel-chromium-molybdenum alloys (Alloys C-4, C-22, and 625), and a titanium-base alloy are ongoing. The experiments were completed that subjected these alloys to brines of various salt content (1 to 10 weight percent NaCl) and pH (2-3, 6-7, and 10-11) at ambient and elevated temperatures (up to 90°C). Results indicate that Alloys 825, G-3, and G-30 underwent pitting and crevice corrosion in all tested environments, with Alloy 825 showing the maximum susceptibility. As to the localized corrosion behavior of nickel-chromium-molybdenum alloys, Alloy C-4 suffered from pitting in all tested environments. But the extent of pitting was less severe than that observed with iron-nickel-chromium-molybdenum alloys. Alloy C-22 and Ti Grade-12 were immune to localized attack under all experimental conditions. Alloy 625 suffered from pitting, crevice and intergranular corrosion in all tested environments under potentiodynamic control.

Galvanic corrosion tests were initiated in January 1997 considering the following relevant factors: (a) environmental factors such as temperature, pH, and electrolytic composition, (b) metallurgical factors such as surface condition and thermomechanical history, and (c) electrode design such as anode-to-cathode area ratio, distance between electrodes, and geometric shapes. These preliminary experiments are currently being performed at ambient temperature using a corrosion-allowance material (A 516) as an anode and a corrosion-resistant alloy (either Alloys 825 and G-3) as a cathode, galvanically connected in an acidic brine by a potentiostat. A516 is compositionally similar to 1020 carbon steel used in other metallic barrier corrosion studies.

As a result of the waste package degradation workshop, some individual models for the different corrosion modes were consolidated, particularly for those modes affecting the inner barrier material and its interaction with the corrosion products and remaining structure of the outer barrier. Work continues on a deterministic model to predict long-term effects of low-temperature oxidation, currently focusing on carbon steel, the principal candidate for the outer barrier container material. Initial emphasis is on humid-air oxidation, in which atmospheric water influences oxidation through condensing on hygroscopic surface contamination or as thin films.

Waste Form Experiments and Modeling

Waste form experiments and modeling included testing and performance evaluations of commercial spent nuclear fuel and defense high-level waste glass dissolution and leaching, spent nuclear fuel oxidation, thermodynamic data development for geochemical modeling, and participation in the waste form degradation and mobilization workshop for developing model abstractions for total system postclosure performance assessment. Summaries of this work follow. Refer to Chapter 5 of this progress report for related work.

Spent nuclear fuel flow-through tests focused in two different areas: (1) uranium oxide (UO_2) matrix flow-through dissolution rate tests on pressurized water reactor and boiling water reactor fuels at a variety of burnups and alkaline and acidic pHs; and (2) gap inventory tests for iodine-129. The four flow-through tests with ATM-105 fuel were completed and three flow-through tests with ATM-103 grain-size powder specimens were started with experimental conditions that complement two ongoing tests. These tests investigate grain boundary leaching rates.

Two commercial spent nuclear fuels, ATM-103 and ATM-106, are being tested for three types of unsaturated conditions: high drip rate tests, low drip rate tests, and vapor tests. A section from the ATM-103 spent nuclear fuel fragment, from the high drip rate test, was examined with scanning electron microscopy after 3.7 years of reaction. Results indicated that reaction occurred primarily as a reaction front through the grains, with limited reaction down the grain boundaries. The depth of reaction was a minimum of 20 μm, the diameter of a totally reacted grain. The transmission electron microscopy examination of another section from the same ATM-103 fragment indicated that (a) technetium, molybdenum, and ruthenium were being removed from epsilon-phase particles in reacted areas of the fuel grains, (b) 1 to 2 weight percent ruthenium and molybdenum were being incorporated into the uranium silicate alteration product present on the surface of the spent nuclear fuel, (c) small amounts (parts per million) of technetium were also incorporated into the uranium silicate alteration product, and (d) plutonium appeared to be concentrating on the fuel surface at areas adjacent to reacted grains. Additional samples and results were obtained for all unsaturated tests after 4.1 years of reaction.

Dynamic light scattering is being developed as a method to study colloids formed from the reaction of spent nuclear fuel and high-level waste glass with ground water under potential repository conditions at Yucca Mountain. The data to be obtained will include size classes and concentrations of colloids present in the solutions. Samples from ongoing waste form corrosion tests were examined as they became available.

Dry-bath weight gain tests are in progress to determine spent nuclear fuel oxidation response. These are long-term tests conducted in a hot cell. These tests primarily use low temperatures (less than 200°C) to examine oxidation rates but one dry bath operates at 255°C to accelerate the oxidation rate. On the basis of information obtained from the dry-bath tests, thermogravimetric apparatus tests were initiated at a higher range of temperatures (250 to

320°C). These two types of tests will provide temperature-time-phase response as UO_2 spent nuclear fuel oxidizes to U_4O_{9+x}, then to U_3O_{8+x}, and finally to UO_3.

The dry-bath oxidation testing continued at a reduced level until January 1997, when a facility-wide electrical outage caused the tests to be shut down. Previous work to interpret the mechanisms of oxidation of spent nuclear fuel from the U_4O_9 ($UO_{2.4}$) state to the U_3O_8 phase has been complicated by extensive scatter in the data from thermogravimetric analysis. Detailed analysis has led to the probability that the scatter is caused by the difference in radial burnup in the samples coupled with the small (200 mg) sample size. This hypothesis seems to be confirmed by analyses of previously oxidized samples of ATM-105 fuel with thermal ionization mass spectrometry. In all instances (within uncertainty) at each temperature, the lower burnup fragments oxidized faster than the higher burnup samples.

Long-term unsaturated tests (drip tests) of two glass compositions (Savannah River Defense Waste Processing Facility and West Valley ATM-10) continued in two test series to determine the types and quantities of radionuclide elements released from waste glasses when subjected to an intermittent dripping water contact scenario. Both soluble and colloidal radionuclide releases of actinides and technetium are being measured. A 304L stainless steel sample holder is also present in these tests to simulate the presence of the pour canister material on glass waste form behavior. As of March 31, 1997, these two series have been in progress for 568 weeks (10.9 years) and 493 weeks (9.5 years), respectively. Both tests were sampled in January. The results are providing data on radionuclide release mechanisms and degradation/alteration rates of glass waste forms for radionuclide release model development.

The thermodynamic data base GEMBOCHS was augmented by the inclusion of reference-state thermodynamic data and heat-capacity coefficients for a large number of cadmium, hafnium, lead, titanium, zinc, and zirconium species, not previously available in GEMBOCHS. These coefficients are used in geochemical modeling software, such as EQ3/6 (Daveler and Wolery, 1992; Wolery, 1992a and b; Wolery and Daveler, 1992).

Development of a radionuclide release rate model for unsaturated conditions, using the unsaturated test data, continued. The model assumes quasi-steady rate processes for dissolution rates, precipitation rates, colloidal kinetics, and adsorption kinetics. On the basis of this assumption, the mass balance equation for a generic species can be simplified and reduced to a mass transport expression that strongly depends on water flux flowing over a wetted area. For unsaturated flows, this results in a weak dependence on the total surface of spent nuclear fuel exposed, in that only the wetted surface contributes to the release rate.

A glass alteration model was developed for use in waste package performance modeling, which is reported in the Waste Form Characterization Report, Version 1.2 (Stout et al., 1996). The model covers water contact modes of trickle flow over the glass surface and gradual immersion in the container in a bathtub mode by a slow inflow of water. The model is based on results of a matrix of batch tests with sudden immersion in a fixed quantity of water and on flow-through tests at relatively high water flow rates. The experimental tests show dependence on temperature, pH, and dissolved silica content in the water. Because the silica content increases with the glass dissolution, the model tracks the changing silica content during the water contact in each mode. A corollary is that the fraction of silica reprecipitated is important in determining the amount of silica remaining in solution.

At the workshop on waste form degradation and radionuclide mobilization, the highest ranked issues for spent nuclear fuel were dissolution/alteration rate, release rate, solubility limits, colloidal kinetics, and cladding degradation. High burnup spent nuclear fuel test samples were also identified as an issue. For high-level waste glass, the highest ranked issues were dissolution/alteration rate, release rate, solubility limits, and colloidal kinetics.

Seal Performance

There was no seal performance assessment activity during this reporting period.

Performance Confirmation

A Performance Confirmation Plan is being prepared, which is based on the performance confirmation concepts study report (CRWMS M&O, 1996z) issued in the previous reporting period and summarized in Progress Report #15. The plan will provide additional details of the planned performance confirmation activities, including surface-based parameter evaluations, evaluations of model predictions, and corrective actions if necessary. As in the concept study report, the plan will (a) identify the processes to be simulated for postclosure performance assessment in support of a license application, (b) list the site and MGDS design parameters needed for these analyses, (c) from this list, recommend the parameters that need to be measured, monitored, observed, tested, and evaluated following the submittal of a license application to construct a repository at Yucca Mountain, and (e) describe specific performance confirmation activities and facilities planned for performance confirmation data acquisition and evaluation.

Site Impact Evaluations

Site impact evaluations are performed to estimate the potential effects of site characterization activities, including the construction and operation of the ESF, on testing and the waste isolation capability of the Yucca Mountain site. Controls on activities are established, if needed, based on these technical analyses to limit adverse effects. Evaluations during this reporting period included an analysis of subsurface water and materials use in the Thermal Testing Facility. The previously established limits for underground water use (CRWMS M&O, 1997x) were reinterpreted to establish specific limits for a new water use activity involving the drilling of a dense pattern of boreholes for instrumentation around the Thermal Testing Facility.

Work concerning the use of committed concrete (i.e., concrete that will remain underground after repository closure) in the subsurface ESF and potential repository is being pursued in an attempt to bound the geochemical effects of such materials on the postclosure waste isolation performance (CRWMS M&O, 1996aa). These effects could include negation of radionuclide sorption in the unsaturated zone, increased neptunium and plutonium concentrations, and earlier and significantly higher radiation doses to future populations. Preliminary recommendations were made with respect to cement composition and concrete production to reduce these effects if concrete liners will be used for the emplacement drifts.

Changes from Previous Progress Reports

This progress report continues the format of Progress Reports started with Progress ,ort #13 (DOE, 1996f) for Chapter 6, which established a one-to-one correspondence with the Site Characterization Plan (SCP) (DOE, 1988) fourth-level performance assessment sections, with the exception of Progress Report Section 6.21 "Site Impact Evaluations," which does not have a corresponding SCP section. As before, not all the SCP sections, however, apply any more to present Yucca Mountain Site Characterization Project (Project) activities. In addition, some performance assessment activities are more logically related to technical subjects covered in other progress report chapters. These instances are called out in the affected sections of this chapter. Section A.5 in Appendix A of this progress report summarizes the SCP approach for performance assessments, describes the current status and changes in these plans, explains the reasons for these changes, and identifies references where these changes and reasons are documented.

6.1 STRATEGY FOR PRECLOSURE PERFORMANCE ASSESSMENT (SCP SECTION 8.3.5.1)

SCP Section 8.3.5.1 addresses the development of the preclosure performance assessment strategy for resolving Key Issue 2. This issue asks whether the projected releases of radioactive materials to restricted and unrestricted areas and the resulting radiation exposures of the general public and workers during repository operation, closure, and decommissioning at Yucca Mountain meet the applicable requirements set forth in 10 CFR Part 20, 10 CFR Part 60, 10 CFR Part 960, and 40 CFR Part 191. In determining the strategy for preclosure performance assessment, due consideration has been given to changes in these regulations.

Part of the effort in this period was directed to producing a draft outline of the Preclosure Radiological Safety Chapter (Chapter 7) of the Project Integrated Safety Assessment. This outline, which is currently under review, was developed to systematically address regulatory requirements. It also includes consideration of radiological safety issues and evaluations that have been required by the NRC in similar licensing activities.

Forecast: Review comments for Chapter 7 of the Project Integrated Safety Assessment will be addressed. Effort will also be directed to producing the appropriate section (preclosure radiological safety) of a technical guidance document for the preparation of the license application, as a continuation of the Chapter 7 work for the Project Integrated Safety Assessment.

6.2 WASTE RETRIEVABILITY (SCP SECTION 8.3.5.2)

SCP Section 8.3.5.2 addresses Issue 2.4, which asks whether the repository can be designed, constructed, operated, closed, and decommissioned so that the option of waste retrieval will be preserved as required by 10 CFR 60.111.

A retrievability requirements study is being conducted during this reporting period and is scheduled for completion at the end of April 1997 (CRWMS M&O, in prep.[g]). The objective of the study is to develop the technical rationale for a decision regarding the MGDS design approach to be used for complying with the 10 CFR Part 60 requirements related to retrievability. Options available for potentially satisfying these requirements range from one extreme of developing a waste package and excavation design that allows retrieval, if necessary, but does not include features (e.g., equipment and surface storage) that facilitate the retrieval, to the opposite extreme of including features in the design that facilitate retrieval and, indeed, building the retrieval capability into the repository at the time of repository development. This activity will recommend the extent to which the repository and waste package designs must accommodate the retrieval option, develop requirements associated with the time required for retrieval, and identify the potential conditions requiring retrieval and the potential conditions involved during waste package retrieval. This study will also develop and evaluate potential scenarios concerning the final disposition of the retrieved waste.

This study will provide the technical bases for retrievability requirements and provide sufficiently specific requirements for MGDS design development, using 10 CFR Part 60 as the basis for developing the requirements. Retrieval will be addressed at the level of detail needed to support development of the license application plan (FY 1997) and the license application itself. The regulatory and licensing organization will use the results in developing the license application plan for FY 1997.

The results of this study will be used by the surface and subsurface design organizations to facilitate the concurrent retrieval design activity. The scope of the design activity is the development of retrieval equipment requirements, capabilities, and descriptions sufficient to support a license application. This activity is developing equipment concepts needed to accomplish the tasks required during the retrieval process. The focus of this activity is the refinement of a reliable retrieval concept and the development of equipment details for retrieval equipment and mechanical components. Waste package retrieval and transport is addressing two conditions: (1) normal waste package retrieval activities (a reversal of waste package emplacement), and (2) off-normal waste package retrieval activities (with specialized retrieval equipment).

Forecast: The Retrievability Strategy Report (CRWMS M&O, in prep.[g]) will be completed in April 1997 and the retrieval design activity will be completed at the end of FY 1997

6.3 PUBLIC RADIOLOGICAL EXPOSURES - NORMAL CONDITIONS (SCP SECTION 8.3.5.3)

SCP Section 8.3.5.3 addresses Issue 2.1, which asks whether expected preclosure radiation doses received by members of the public during repository operation, closure, and decommissioning will be less than allowed by 10 CFR 60.111, 40 CFR 191 Subpart A, and 10 CFR Part 20

See Section 4.2 of this progress report

6.3.1 Performance Assessment Activity 2.1.1.1 - Refinement of Site Data Parameters Required for Issue 2.1

The objective of this activity is to refine the population, agricultural, surface water, meteorological, host rock, and offsite nuclear installation data needed for determining preclosure radiological exposures to members of the public resulting from normal repository operations. This information is collected as part of the environmental, geological, and hydrological site programs described in Chapter 3 of this progress report.

The ongoing meteorological program (as reported in detail in Section 3.8 of this progress report) has collected and analyzed additional wind data that allow more accurate predictions of dispersion and extreme conditions. Commercially available software is being evaluated that would allow local topography to be considered in dispersion calculations.

A telephone study of the residential population in the vicinity of the proposed facility was conducted. This study was undertaken to define the population in terms of parameters needed to assess the chronic effects of radionuclide releases in both the preclosure and postclosure phases of the potential repository.

Forecast: Collection and analysis of required site specific data will continue.

6.3.2 Performance Assessment Activity 2.1.1.3 - Advanced Conceptual Design Assessment of the Public Radiological Safety during the Normal Operations of the Potential Yucca Mountain Repository

The objective of this activity is to perform a public radiological safety assessment of the advanced conceptual design of a potential Yucca Mountain repository. Secondary objectives are to provide information for the refinement of the site data parameters list for SCP Issue 2.1 (Performance Assessment Activity 2.1.1.1, see Section 6.3.1 of this progress report) and to provide feedback to the preclosure risk assessment methodology program for future methods development activities (Performance Assessment Activity 2.1.1.2, see Section 6.3.3 of this progress report).

See Section 4.2 of this progress report.

6.3.3 Performance Assessment Activity 2.1.1.2 - Development of Performance Assessment Activities through the Preclosure Risk Assessment Methodology Program

The objective of this activity is to benefit from the performance assessment methods development efforts for the preclosure risk assessment methodology program. A secondary objective is to use the information developed in this activity to assist in refining the site parameters list for SCP Issue 2.1 (Performance Assessment Activity 2.1.1.1, see Section 6.3.1 of this progress report).

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.4 WORKER RADIOLOGICAL SAFETY - NORMAL CONDITIONS (SCP SECTION 8.3.5.4)

SCP Section 8.3.5.4 addresses Issue 2.2, which asks whether the repository can be designed, constructed, operated, closed, and decommissioned in a manner that ensures the preclosure radiological safety of workers under normal operations as required by 10 CFR 60.111 and 10 CFR Part 20. Results of analyses are reported in Section 4.2 of this progress report.

6.4.1 Performance Assessment Activities 2.2.1.1 and 2.2.2.1 - Refinement of Site Data Parameters Required for Issue 2.2

The objective of this activity is to refine (a) the data needed on the subsurface radiation environment due to natural and man-made radioactivity and (b) the meteorological, host rock, and ground-water data needed for determining radiological exposures to workers resulting from normal repository operations. This information is collected as part of the environmental, geological, and hydrological site programs described in Chapter 3 of this progress report.

Forecast: No performance assessment work is planned for the remainder of FY 1997.

6.4.2 Performance Assessment Activities 2.2.1.2 and 2.2.2.3 - Advanced Conceptual Design Assessment of the Worker Radiological Safety during the Normal Operations of the Potential Yucca Mountain Repository

The objective of this activity is to perform a worker radiological safety assessment of the advanced conceptual design for a potential Yucca Mountain repository. Secondary objectives are to provide information for the refinement of the site data parameters list for SCP Issue 2.2 (Performance Assessment Activities 2.2.1.1 and 2.2.2.1, see Section 6.4.1 of this progress report) and to provide feedback to the preclosure risk assessment methodology program for future methods development activities (Performance Assessment Activity 2.2.2.2, see Section 6.4.3 of this progress report).

This activity has been conducted as an ongoing design activity and is reported in Section 4.2.1 of this progress report.

Forecast: This will be a continuing design activity.

6.4.3 Performance Assessment Activity 2.2.2.2 - Development of Performance Assessment Activities through the Preclosure Risk Assessment Methodology Program

The objective of this activity is the development of performance assessment activities to benefit from the preclosure risk assessment methodology program. A secondary objective is to use the information developed in this activity to assist in refining the site data parameters list for SCP Issue 2.2 (Performance Assessment Activities 2.2.1.1 and 2.2.2.1, see Section 6.4.1 of this progress report).

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.5 ACCIDENTAL RADIOLOGICAL RELEASES (SCP SECTION 8.3.5.5)

SCP Section 8.3.5.5 addresses Issue 2.3, which asks whether the repository can be designed, constructed, operated, closed, and decommissioned in such a way that credible accidents do not result in projected radiological exposures of the general public and of workers in excess of applicable limiting values.

See Section 4.2 of this progress report.

6.5.1 Performance Assessment Activities 2.3.1.1 and 2.3.2.1 - Refinement of Site Data Parameters Required for Issue 2.3

The objective of this activity is to refine the population, agricultural, surface-water, and meteorological data needed (a) for determining credible accident sequences and their respective frequencies, (b) for developing candidate design basis accidents, and (c) for determining preclosure radiological exposures to members of the public and to workers as a result of credible accidental radiological releases. This information is collected as part of the environmental, geological, and hydrological site programs described in Chapter 3 of this progress report.

The only item identified as a potential problem in the last progress report (maximum wind speed) has been studied. See Section 6.3.1 of this progress report for details.

Forecast: See Section 6.3.1 of this progress report.

6.5.2 Performance Assessment Activity 2.3.1.2 - Determination of Credible Accident Sequences and their Frequencies Applicable to the Potential Yucca Mountain Repository

The objective of this activity is to develop a comprehensive list of accidents that are both credible and applicable to a potential Yucca Mountain repository.

As reported in the previous progress report and again discussed in Section 4.2 of this progress report, the Preliminary Hazards Analysis (CRWMS M&O, 1996q) has been used as source document for event definition. As design details emerge for each of the waste handling processes, event probabilities are assessed.

6.5.3 Performance Assessment Activity 2.3.1.3 - Development of Candidate Design Basis Accidents for the Potential Yucca Mountain Repository

The objective of this activity is to develop a set of candidate design basis accidents to be analyzed as part of the total safety analysis.

See Sections 4.2 and 6.5.2 of this progress report.

Forecast: See forecast at the beginning of Section 6.5 of this progress report.

6.5.4 Performance Assessment Activity 2.3.2.2 - Consequence Analyses of Credible Accidents at the Potential Yucca Mountain Repository

The objective of this activity is to determine the consequences of credible accidents in terms of radiation doses to the essential repository workers and the public.

See Section 4.2 of this progress report.

6.5.5 Performance Assessment Activity 2.3.2.3 - Sensitivity and Importance Analyses of Credible Accidents at the Potential Yucca Mountain Repository

The objectives of this activity are (a) to quantify uncertainties and sensitivities in the accident risk assessment and (b) to establish importance rankings for systems, structures, and components of a potential Yucca Mountain repository with respect to radiological safety.

See Section 4.2 of this progress report.

6.5.6 Performance Assessment Activity 2.3.2.4 - Documentation of Results of Safety Analyses and Comparison to Applicable "Limiting" Values

The objectives of this activity are (a) to produce documentation of the results of the accident risk assessment in the necessary format and (b) to make comparisons of the results to applicable limiting values. This activity will complete the resolution of SCP Issue 2.3 at the end of the license application design.

See Section 4.2 of this progress report.

6.6 HIGHER-LEVEL FINDINGS - PRECLOSURE RADIOLOGICAL SAFETY (SCP SECTION 8.3.5.6)

SCP Section 8.3.5.6 addresses Issue 2.5, that asks whether the higher-level findings required by 10 CFR Part 960 can be made for the qualifying condition of the system guideline and the qualifying and disqualifying conditions of the technical guidelines for population density and distribution, site ownership and control, meteorology, and offsite installations and operations.

No progress was made during the reporting period; this was an out-year activity. See Section 2.2.1 of this progress report for relevant regulatory activities.

Forecast: No performance assessment work is planned for FY 1997.

6.7 HIGHER-LEVEL FINDINGS - EASE AND COST OF CONSTRUCTION (SCP SECTION 8.3.5.7)

SCP Section 8.3.5.7 addresses Issue 1.4, that asks whether the higher-level findings required by 10 CFR Part 960 can be made for the qualifying condition of the system guideline and the qualifying and disqualifying conditions of the technical guidelines for surface characteristics, rock characteristics, hydrology, and tectonics.

No progress was made during the reporting period; this was an out-year activity. See Section 2.2.1 of this progress report for relevant regulatory activities.

Forecast: No performance assessment work is planned for FY 1997.

6.8 STRATEGY FOR POSTCLOSURE PERFORMANCE ASSESSMENT (SCP SECTION 8.3.5.8)

SCP Section 8.3.5.8 addresses the development of the postclosure performance assessment strategy for resolving Key Issue 1. This issue asks whether the MGDS at Yucca Mountain will isolate the radioactive waste from the accessible environment after closure in accordance with the requirements of 10 CFR Part 60, 10 CFR Part 960, and 40 CFR Part 191. Postclosure performance assessment activities have been revised in anticipation of revised U.S. Environmental Protection Agency standards for Yucca Mountain as mandated in the Energy Policy Act of 1992. This Act mandates a process for setting a standard to be applied specifically to the potential repository system at Yucca Mountain. The resultant changes in the performance assessment activities primarily result from the expected change from a release-based to a dose-based standard. Use of a dose-based standard will require inclusion of the biosphere and additional emphasis on elements of the saturated zone.

The viability assessment for the potential repository system, as defined in the FY 1997 Energy and Water Appropriations Act, includes four major components: three of the components are related to (1) repository and waste package design, (2) plan and cost to complete a license application, and (3) an estimate of costs to construct the repository. The fourth component is "a total system performance assessment based on the design concept and 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." Part of the basis for developing the total system performance assessment is the subject of a series of abstraction-testing activities described in the following sections.

The models used to perform the total system performance assessment for the viability assessment are generally expected to be formulated as "ab ons" from more detailed process models. For a total system performance assessment, an abstraction is defined as a simplified or idealized model that reproduces or bounds the essential elements of a more detailed process model. For an abstraction, the inputs may be those that form a subset of those required for a process model, or they may be a response function derived from intermediate results. Regardless, the abstracted form must capture uncertainty and variability. The abstractions must also be tested against process models to ensure their validity. Abstractions are used because of th probabilistic or stochastic nature of total system performance assessment analyses. The intent of the abstraction process is to retain key aspects of process models, while producing results usable in multiple realization probabilistic models.

Following is a general di ssion of the activities currently ongoing to produce abstracted models. Also presented are the cific results generated to date for the activities that have been initiated.

6.8.1 Abstraction-Testing Activities

During FY 1997, a series of abstraction testing activities were initiated to identify and construct appropriate numerical or analytical representations of components of the potential Yucca Mountain repository system to ensure the development of a valid, defensible total system performance assessment for the viability assessment. This objective requires that performance assessment incorporate the most complete and current information available from the Project. The objective also requires that the essential behavior of key processes (defined relative to the contribution that each process makes to long-term performance of the repository system) of each component be identified and captured in a computationally efficient manner. The important issues, including the alternative hypotheses, must be identified, quantified, and evaluated. Because of time and resource constraints, the model development must be focused on only those issues that are most important to performance. And, to provide traceability and transparency, the bases for assumptions must be well defined, justified, and documented.

The total system performance assessment to be performed for the viability assessment will be constructed of models developed to represent processes and features of both the natural and the engineered barrier system. Although the responses of the components are strongly interdependent, the performance assessment analysts have broken the processes into somewhat artificial components to facilitate analysis. The nine components are: (1) unsaturated zone flow, (2) waste package degradation, (3) unsaturated zone thermohydrologic flow, (4) unsaturated zone transport, (5) waste form alteration and mobilization, (6) near-field environment, (7) potential nuclear criticality, (8) saturated zone flow and transport, and (9) biosphere. A separate abstraction-testing activity has been defined for each of these nine components.

To meet the goal of constructing a valid, defensible total system performance assessment for the viability assessment, the abstraction-testing activities were designed to integrate site characterization, MGDS design, environmental programs, and performance assessment. To achieve this integration, analysts from each of these areas of the project have been identified to participate in all aspects of the activities. These activities have three major elements. The first part includes the planning needed to identify a preliminary list of relevant issues for the subject component and to define the activities to be accomplished in a workshop. This work is accomplished by a team (the Abstraction Core Team) that includes at least one subject matter expert in the component of interest, a total system performance assessment expert, and a performance assessment subsystem modeler. The performance assessment subsystem modeler is the task lead for the entire activity. The next step in the activity is to hold a workshop to develop a consensus on the relative importance of issues related to the primary process and to develop plans to analyze the highest ranked issues. The schedule for the workshops and for the deliverables reporting the results of the workshop is shown in Table 6-1. The last stage is the implementation of the analyses identified during the workshop process to develop the parameters, models of processes, and alternate conceptualizations for use in the total system performance assessment for the viability assessment.

Because the ultimate goal of these activities is a single total system performance assessment for the viability assessment, there is a need to integrate the components for the final analyses. The primary responsibility for the integration process lies with an oversight group called the Total System Performance Assessment Core Team and with performance assessment management. The Total System Performance Assessment Core Team and performance assessment management attend all the workshops and a representative from the Total System Performance Assessment Core Team is part of each Abstraction Core Team to ensure consistency and usefulness of the products generated by all the activities. In addition, the Abstraction Core Team leads are responsible for ensuring consistency and commonality among the various inputs, outputs, and model domains for all the analyses. The leads for unsaturated zone flow, unsaturated zone transport, and thermohydrology met in February 1997, to develop a schedule that logically linked all their activities and analysis "deliverables." They also determined the need to use a single "base case" for certain analyses and assigned responsibilities

Table 6-1. Performance Assessment Abstraction Testing Workshops

Topic	Workshop Dates	Report Deliverable Date
Unsaturated Zone Flow	12/11/96 - 12/13/96	02/27/97
Waste Package Degradation	01/08/97 - 01/10/97	02/24/97
Unsaturated Zone Thermohydrology	01/21/97 - 01/23/97	03/18/97
Unsaturated Zone Transport	02/05/97 - 02/07/97	04/30/97
Waste Form Alteration and Mobilization	02/19/97 - 02/21/97	05/15/97 & 06/25/97
Near-Field Environment	03/05/97 - 03/07/97	06/30/97
Nuclear Criticality	03/18/97 - 03/20/97	09/25/97
Saturated Zone Flow and Transport	04/01/97 - 04/03/97	06/30/97
Biosphere	06/03/97 - 06/05/97	08/15/97

for establishing this base case. A similar meeting for the Abstraction Core Team leads associated with the engineered components will meet in early April after the criticality workshop.

6.8.2 Workshop Process Synopsis

In a very general sense, all the abstraction-testing workshops can be broken down into three phases: pre-workshop planning, workshop implementation, and post-workshop follow-up. Although each workshop will vary somewhat from other workshops, each follows a basic format described in the following paragraphs.

During the pre-workshop planning, the Abstraction Core Team, along with their facilitator, (a) defines the scope of the workshop in terms of issues (i.e., important parameters, processes, and alternative conceptual models) relevant to the workshop topic (unsaturated zone flow, unsaturated zone thermohydrology flow, waste package degradation, etc.); (b) defines the workshop participants list and elicits input from them; (c) determines appropriate prioritization criteria to rank topic issues from the relevant subsystem performance measures; (d) revises the agenda using participant inputs; (e) creates an initial list of subissues expected to be developed at the workshop; and (f) assigns each participant to one of four working groups to include data collectors, process modelers, subsystem modelers, and total system performance assessment modelers.

The workshop itself is a facilitated activity that has three individual goals: (1) to develop a comprehensive list of issues that might be addressed in analyses after the workshop; (2) to prioritize the subissues to ensure that most of the effort in the abstraction-testing activity is dedicated to those issues expected to have the most impact on long-term performance; and (3) to develop analysis plans to address the high-priority issues.

Following each workshop, the primary task for the post-workshop followup is completing the details of the abstraction-testing plans. These, along with the details related to the workshop planning and implementation, are incorporated into a deliverable report.

The results of the workshops held to date are summarized below. First is a listing of the performance measures or prioritization criteria against which all issues were ranked for importance to postclosure performance. Next is a list of the highest priority issues, as defined during the workshop. The number of issues in each list is variable and was based on where there appeared to be significant "breaks" in the numerical values derived for each during the prioritization process. Finally, a short synopsis of the current expectations for the various analysis plans is shown. These results are preliminary and are expected to evolve as analyses proceed and new information is obtained. The wording of the criteria, the issues list, and the analysis plan summaries are taken directly from the workshops, and so the style of presentation and wording varies from group to group.

6.8.3 Unsaturated Zone Flow Abstraction-Testing Workshop Results

Criteria for Prioritization

- Does the issue have a strong effect on percolation flux at the repository?
- Does the issue have a strong effect on seepage into the drift?
- Will the issue be important to flow and transport below the repository?
- Does the issue have a strong effect on the partitioning of flow between the fractures and the matrix?

Top Priority Issues for Unsaturated Zone Flow

1. Issues related to infiltration:
    - Spatial variability resolution
    - Temporal variability resolution
    - Range of values in the infiltration model (uncertainty)
    - Appropriate inclusion of climate change.

2. Issues related to lateral flow:

   - Does lateral flow contribute to focusing flow?

   - What is the appropriate dimensionality to use to model lateral flow for total system performance assessment?

   - What are the potential impacts of hydrothermal alteration on lateral flow?

   - How do the properties of zeolites affect lateral flow (including fracture versus matrix flow)?

3. Issues related to perched water:

   - How are perched water bodies formed?
   - What is the source for perched water bodies?
   - How representative are water chemistry and isotopic data?
   - What is the extent of the perched water bodies?
   - How do thermal perturbations affect perching?
   - How is perched water considered in calibration?

4. Issues related to fracture matrix interaction:

   - Can direct correlation between fracture matrix coupling and fracture saturation be assumed?

   - What are the differences in fracture/matrix coupling in different hydrogeologic units and faults?

   - What features processes, and parameters affect fracture/matrix interactions (coatings, connectivity, aperture, etc.)?

   - How does fracture/matrix interaction change with infiltration changes?

5 and 6. Issues related to flow channeling and seepage into the drifts:

   - How do thermal/mechanical effects change channeling and seepage?

   - Are seep locations predictable?

   - How do fracture/matrix properties impact seeps and channeling?

   - How do different conceptual models impact seepage and channeling?

   - How do flux differences influence seeps and channeling (spatial, temporal, volume)?

7. Issues related to matrix properties:

   - How should matrix properties be upscaled?
   - Should grid-scale heterogeneity be included?
   - How should correlated parameters be treated?
   - Should subgrid block fractures be lumped with the matrix?

8. Issues related to fracture properties:

   - Are bulk hydraulic conductivities values representative?
   - What are the appropriate conceptual models for fracture flow?
   - How should spatial heterogeneity of fractures be considered?
   - How should discrete fractures be scaled?
   - Are bulk hydraulic conductivities related to other fracture properties?

9. Issues related to calibration:

   - What are the applicability and robustness of available data to calibration?

   - What is the appropriate approach for model calibration?

   - Should models be "calibrated" or "bounded"? Is it different for different parameters?

   - Should faults be part of the calibration?

10. Issues related to conceptual models:

   - Should a hybrid model for total system performance assessment be developed (dual permeability model, equivalent continuum model, Weeps model)?

   - Should flow for total system performance assessment be modeled in three dimensions, two dimensions, or one dimension?

   - How should spatial variability in the parameters be considered?

   - What is the appropriateness of transient versus steady state?

   - How should different models for thermohydrology, flow, and transport be dealt with?

   - Can time-dependent total system performance assessment calculations be done? Are different models for "hot" versus "cool" periods needed?

   - Can changes in flow paths be constrained with changes in parameters?

Synopsis of Analysis Plans for Addressing Top-Ranked Unsaturated Zone Flow Issues

Analysis Plan 1: Sensitivity Studies Conducted on the Site-Scale Model to Determine Abstraction Methods for Unsaturated Zone Flow

The objective of this plan is to produce a simplified model of the unsaturated zone from which numerous simulations can be run for unsaturated zone abstractions for total system performance assessment and also to conduct sensitivity studies to help prioritize and clarify related issues.

Analysis Plan 2: Flow Seepage into Drifts under Pre-Waste-Emplacement Conditions

The objective of this plan is to develop a drift-scale model of seepage into drifts for total system performance assessment. This model will specifically address the spacing of the drips and under what hydrogeological conditions water will drip into the drifts.

Analysis Plan 3: Testing of Perched-Water Concepts and their Implications for Total System Performance Assessment Calculations for the Viability Assessment

The objective of this plan is to identify physical controls on perched-water formation and to test assumptions through numerical simulation using two-dimensional and three-dimensional models. The assumptions to be tested are that the infiltration spatial distribution, pump test data, and geochemical signature of the perched water body are important to understanding the location and extent of perched water and that the conceptual model of the formation of the perched water plays a key role in understanding the volume and residence times of the perched water bodies.

Analysis Plan 4: Subgrid-Scale Fractures and Model Calibration

The objective of this plan is to determine the sensitivity of subgrid scale fractures to radionuclide transport calculations in order to simplify total system performance assessment calculations.

6.8.4 Waste Package Degradation Abstraction/Testing Workshop Results

Criteria for Prioritization

- How significantly does the process/issue affect the time of waste package failure?
- How significantly does the process/issue affect the rate of waste package failure?
- How significantly does the process/issue affect the rate of waste package perforation, and thus the rate of radionuclide release from the waste package?

Top Priority Issues ' Waste Package Degradation

1. Issues related to outer barrier corrosion:

   - Refluxing and concentration of electrolytes
   - Microbiological conditions (aerobic/anaerobic)
   - Temperature dependence on corrosion
   - Model of salt buildup
   - Critical relative humidity (dry-humid)
   - Critical relative humidity (humid-aqueous)
   - Aqueous corrosion (localized/pitting)
   - Flow rate and episodicity of water.

2. Issues related to inner barrier corrosion:

   - Aqueous corrosion (localized/pitting)
   - Crevice corrosion
   - Cathcdic protection
   - Choice of waste package materials
   - Barrier interface environment.

3. Issues related to galvanic effects

   - Barrier materials (alloy choice)
   - Water chemistry versus time
   - Crevice corrosion (including at welds)
   - Threshold for galvanic protection cessation
   - Ionic conductivity at interface
   - Electrode area ration
   - Fabrication process (contact effectiveness)
   - Water-contact mode inside and outside container
   - Negative effects of ferric ions on the inner barrier.

4. Issues related to microbially induced corrosion:

   - Water availability
   - Amount of nutrients
   - Susceptibility of inner barrier
   - Preferential weld susceptibility
   - Container materials (microconstituents).

5. Issues related to rockfall and juvenile failures:

   - Timing of rockfall
   - Backfill (design or natural)
   - Time dependency of thinning of waste package walls (including structural failure)
   - Drift size (rockfall impact).

Synopsis of Analysis Plans for Addressing Top-Ranked Waste Package Degradation Issues

Analysis Plan 1: Carbon-Steel Outer-Barrier Corrosion

The four objectives of this plan are to: (1) develop a model of humid-air general corrosion for the carbon-steel outer barrier for use in the waste-package degradation model for total system performance assessment; (2) develop a model of aqueous general corrosion for the outer barrier, including the transition from non-aqueous to aqueous processes, for use in the waste-package degradation model for total system performance assessment; (3) develop a model to represent localized corrosion (or variation in corrosion) of the outer barrier in humid-air and aqueous corrosion conditions; and (4) exercise the models to investigate the sensitivity of waste package degradation to the corrosion of the carbon-steel outer barrier.

The models and abstractions will be developed based on the following three hypotheses: (1) humid air corrosion can be represented as a function of relative humidity, temperature, salt scale formation, and water dripping; (2) aqueous corrosion can be represented as a function of pH, temperature, water chemistry, relative humidity, and water contact duration; and (3) localized variations in corrosion on a single waste package can be represented by a pitting factor as a multiplier on the (average) general corrosion depth. The multiplier may vary as a function of corrosion depth.

Analysis Plan 2: Corrosion-Resistant Inner-Barrier Corrosion

The objective of this plan is to develop a corrosion model for predicting the rate of penetration of the inner barrier, which consists of corrosion-resistant material, as a function of the near-field environment. The near-field environment is characterized by temperature, humidity, in-drift water dripping, and the chemistry of the contacting water. Penetration of the corrosion resistant material will be assumed to be caused by localized corrosion (i.e., pitting and active crevice corrosion). This modeling activity accounts for the interaction between the outer barrier, which consists of a corrosion-allowance material, and the inner barrier. Interactions will include (a) pH suppression in the crevice caused by the hydrolysis of products from corrosion-allowance material corrosion, (b) crevice formation between those precipitates and the corrosion-resistant material, (c) galvanic coupling, and (d) the accumulation of corrosion products. Several of these effects will be accounted for with a near-field environment correction (calculation of pH and mixed potential) applied at the interface between the corrosion-allowance material and corrosion-resistant material. Microbial action, such as the conversion of Fe(II) to Fe(III), can be considered in this interfacial near-field environment correction.

The analysis will be based on several hypotheses. As the outer barrier degrades, the inner barrier will be exposed in patches. Penetration of the corrosion-allowance material will be by either (a) humid air corrosion or (b) aqueous corrosion. Each exposed area (or patch) can be subdivided into three generic zones. Zone 1: the corrosion-resistant material will be directly exposed to the near-field environment, via humid air or a thin layer of oxygenated and acidified water. Zone 2: the corrosion-resistant material will be exposed to a thin layer of acidified water, with a gradient in oxygen concentration. Zone 3: the corrosion-resistant material will be exposed to a thin layer of acidified and deoxygenated water.

Analysis Plan 3. Microbiologically Influenced Corrosion

The three objectives of this plan are to: (1) develop the best model(s) possible in the time available for total system performance assessment for the viability assessment; (2) identify sources of information that can be acquired to test the model(s) (i.e., literature, laboratory testing, natural analogs); and (3) exercise the model(s) and present the results to a body of experts.

It is assumed that microbiologically influenced corrosion can be modeled as localized corrosion incorporating additional factors such as temperature, water availability, nutrient availability, and pH.

Analysis Plan 4: Effects of Variability in Near-Field Conditions, Manufacturing, and Materials on Waste Package Degradation

The three objectives of this plan are to: (1) develop model(s)/abstractions(s) to represent variability in waste-package materials, waste-package manufacturing, and near-field conditions including rockfall; (2) develop method(s) to incorporate the model(s)/abstraction(s) for the variabilities into the waste-package degradation model; and (3) exercise the model(s)/ abstraction(s) to investigate the sensitivity of waste-package degradation to these variabilities.

The models and abstractions will be developed based on the following four hypotheses: (1) effects of variability in waste package materials, waste-package manufacturing, and near-field conditions including rockfall can be represented by sampling over individual model parameters; (2) there is a physical basis for a localization factor to represent enhanced corrosion at the welded regions of the carbon-steel outer container; (3) enhanced corrosion at the welded regions of the corrosion-resistant inner container can be represented by changes in the corrosion model parameters; and (4) effects of rockfall/backfill on the waste-container corrosion processes can be represented as providing preferential sites for localized corrosion processes. Rockfall/backfill would form crevices where it contacts the waste packages and provide wetter conditions at these contact points.

6.8.5 Unsaturated Zone Radionuclide Transport Abstraction-Testing Workshop Results

Criteria for Prioritization

- Radionuclide concentration
- Water flux
- Radionuclide velocity
- Temporal and spatial distribution of travel time to the water table (i. e., spread of the breakthrough curve.

Top Priority Issues for Unsaturated Zone Radionuclide Transport

1. Issues related to physical transport processes:

   - What conceptual model should be used for fracture/matrix interactions?
   - How should long-term transient flow be included in unsaturated zone radionuclide transport modeling?
   - What range and dependencies should be used for the fracture/matrix interaction parameters?
   - What are key fracture and matrix properties to consider (i.e., fracture porosity)?

2. Issues related to chemical interactions and repository perturbed environment:

   - Is the minimum Kd approach an appropriate modeling approach for unsaturated zone radionuclide transport?
   - Do colloids play an important role in unsaturated zone radionuclide transport?
   - Is thermal-chemical alteration of existing minerals important for unsaturated zone radionuclide transport?

3. Issues related to heterogeneity and model calibration:

   - Is lateral diversion of radionuclide pathways important for unsaturated zone radionuclide transport?
   - Is a more detailed stratigraphy than currently modeled (Topopah Spring welded, Topopah Spring vitric, Calico Hills nonwelded vitric, Calico Hills nonwelded zeolitized, and Prow Pass nonwelded tuff hydrogeologic unit) below the repository important for unsaturated zone radionuclide transport?
   - Are areal variations in abundance and composition of zeolites to be important for unsaturated zone radionuclide transport?

Synopsis of Analysis Plans for Addressing Top-Ranked Unsaturated Zone Transport Issues

Analysis Plan 1: Fracture/Matrix Interaction

The objective of this plan is to conduct sensitivity studies investigating the effects of fracture/matrix interaction on unsaturated zone radionuclide transport. These studies will identify the significance of fracture/matrix interaction for unsaturated zone transport under differing transport conditions and which parameter ranges are important for a total system performance assessment. The parameter sensitivities to be investigated are the matrix sorption coefficient, fracture sorption coefficient, and matrix diffusion coefficient for each unit.

Variations in the flow field will be investigated through variations in infiltration and the fracture/matrix interaction parameter. Comparisons between three-dimensional and two-dimensional models will be made for base-case infiltration and the fracture/matrix interaction parameter to help calibrate interpretations of results from two-dimensional models for the real system. Breakthrough curves and peak mass flux at the water table will be used to compare the results of different transport calculations. Relationships between the chemical transport parameters, peak mass flux value, and the time to peak mass will be developed. This may be done in the form of a response surface between the five parameters and the peak mass flux value and time of peak mass flux arrival.

Analysis Plan 2: Transient Flow and Transport

The objective of this plan is to consider an abstraction approach for treating the effects of longer-term transient flow and radionuclide transport using a quasi-steady flow and transport calculation. This analysis will be carried out using the three-dimensional site-scale model for simulations of unsaturated zone flow and radionuclide transport with longer-term transient flow resulting from the effects of climate change on infiltration. The calculations will be performed using a dual permeability model without the effects of repository heating. A spatially variable infiltration rate that is scaled temporally will be used as the upper boundary condition. Sensitivities will be evaluated by varying the base infiltration rate between 1 and 10 mm/year. The simulations will be run for 10,000 years. A quasi-steady flow field that tracks changes in infiltration rate will be tested as a model abstraction for simplifying and bounding the effects of fully transient flow and transport. The lateral boundaries will be modeled as impermeable to flow and transport. The lower boundary is defined by the water table. Both the present water table and 100 m above the present water table (wetter future climates) will be used. The ranges of radionuclide release rates from the potential repository will be based on Total System Performance Assessment - 1995 analyses (CRWMS M&O, 1995e), unless additional information is available. The calculations will consider sorbing (neptunium-237) and nonsorbing (technetium-99) radionuclides.

Analysis Plan 3: Colloid-Facilitated Radionuclide Transport

The objective of this plan is to assess the role of colloids in facilitating radionuclide transport, and if significant, attempt to provide an abstracted model for total system performance assessment. The subject of colloid transport will be studied for plutonium colloids. Two flow/transport models will be tested: (1) a one-dimensional calculation in which fracture transport of colloids is unaffected by matrix interactions; and (2) a detailed, two-dimensional transport calculation. The detailed flow and transport models will use base-case hydrogeologic parameters and boundary conditions.

Analysis Plan 4: Sorption Models for Radionuclide Transport

The objective of this plan is to assess the effects of using Kds versus more sophisticated sorption models on radionuclide transport through the unsaturated zone. However, more complete documentation is required to support the model abstraction that a linear Kd model bounds the effects of more complex chemical interactions between radionuclides and rock that are known to occur. Therefore, existing work will be reviewed and summarized to see if further analysis on this subject is warranted.

Analysis Plan 5: Effects of Dispersion and Fine-Scale Heterogeneity on Radionuclide Transport

The objective of this plan is to test the impact of fine-scale heterogeneous mineral distributions and physical dispersion on models of radionuclide transport. This modeling effort will help define the relative importance of these fine-scale features, the use of effective properties, and physical dispersion on unsaturated zone radionuclide transport. Flow and transport calculations will use the base-case unsaturated zone flow parameters and boundary conditions and base-case transport properties in a two-dimensional model domain. Higher-resolution gridding will be used to capture fine-scale heterogeneity and to more accurately represent physical dispersion. Heterogeneous property distributions will be derived from information available from the three-dimensional mineralogic model (Chipera et al., in prep.) combined with geostatistical realizations. The model will be used to simulate transport for nonsorbing (technetium-99) and poorly sorbing (neptunium-237) species. The radionuclide source term will be spatially distributed throughout the potential waste emplacement drifts represented in the model. The sensitivity calculations will be performed for cases in which the radionuclide inventory is released over 1000, 10,000, and 100,000 years. Breakthrough curves for conservative (technetium-99) and poorly sorbing (neptunium-237) radionuclides can be used to distinguish the effects on radionuclide transport. Comparisons with calculations using a coarse-gridded model will be used to test the use of effective parameters and the influence of physical dispersion. The distribution of mean travel time for alternative representations of the heterogeneous case will be compared with each other and a homogeneous stratigraphic case.

Analysis Plan 6: Use of Environmental Data for Unsaturated Zone Flow and Radionuclide Transport

The two objectives of this plan are to: (1) ensure that total system performance assessment modelers are aware of the existing data bases that bear on flow and transport issues, and (2) provide process modelers and abstraction modelers with a status report that attempts to integrate the various geochemical and isotopic lines of evidence. The final product will be a report that synthesizes the data for each issue, clearly identifying what conclusions can be considered "firm," and what aspects are considered inconsistent, inconclusive, or inappropriate (e.g., because of unreliable data or questionable assumptions). This report will include an assessment as to whether the evidence supports or refutes various conceptual models of flow and transport for the Yucca Mountain site and whether these models would be provided within a time frame that allows such information to be useful for influencing the total system performance assessment for the viability assessment. A list of key performance assessment issues that can be addressed by environmental data, beginning with the lists of issues prepared for the two workshops on unsaturated zone flow and transport will be identified.

6.8.6 Waste Form Alteration and Mobilization Abstraction-Testing Workshop Results

Criteria for Prioritization

- Radionuclide concentration at the waste form
- Mass release rate of radionuclides from the engineered barrier system
- Time and spatial variability in mass release rate
- Form of radionuclides entering the unsaturated zone for transport.

Top Priority Issues for Waste Form Alteration and Mobilization

1. Issues related to spent nuclear fuel:

    - Dissolution rate
    - Time dependent evolution of solution and alteration layer
    - Representation of evolution of the near field.

2. Issues related to defense high-level waste and other spent nuclear fuel types:

    - Time dependent evolution of solution and alteration layer
    - Vapor hydration
    - Evolution of near-field environment
    - Dissolution rate.

3. Issues related to mobilization and transport:

    - Physical processes - water contact mode
    - Colloids
    - Chemical processes - mobilization - fluid dependence
    - Physical processes - transport paths.

Synopsis of Analysis Plans for Addressing Top-Ranked Waste Form Alteration and Mobilization Issues

Analysis Plan 1: Cladding and Canister Credit

The objective of this plan is to develop a time-dependent distribution for cladding and/or canister perforation and fuel exposure. The activity will propose how to take credit for and model the performance of cladding and/or canisters. This model will be developed for commercial spent nuclear fuel cladding and may be extended to U.S. Department of Energy (DOE)-owned spent nuclear fuels and canistered waste.

Analysis Plan 2: Spent Nuclear Fuel Dissolution

The objective of this plan is to develop a time-dependent description of spent nuclear fuel surface area and dissolution that will provide fuel-dominated water chemistry and fuel dissolution rate (to Analysis Plan 3 below). Input to this activity will include the initial spent nuclear fuel condition and the chemical composition of the inflowing water. This activity will produce an updated model for oxide fuel alteration and dissolution consistent with current experimental results.

Analysis Plan 3: Post-Dissolution Water Chemistry and Precipitated Phase Formation

The objective of this plan is to take dissolution model output (from Analysis Plan 2 above), including water chemistry, to determine rate of precipitated phase formation (secondary phases). The output includes (a) dissolved and transportable species (colloids) that provide radionuclide release rate from the waste form and (b) altered water chemistry for further waste form interactions. This activity will provide an improved representation of the chemical processes at the waste form surface that control the mobilization of radionuclides.

Analysis Plan 4: Defense High-Level Waste Glass Degradation and Radionuclide Release

The objective of this plan is to model the alteration of defense high-level waste glass and the release of radionuclides as a function of temperature, water chemistry, water contact mode and the extent of vapor hydration before liquid water contact. This activity will produce an improved model for standard defense high-level waste glass.

Analysis Plan 5: Solubility Limits on Dissolved Radionuclides

The objective of this plan is to derive constraints on dissolved radionuclide concentrations based on the long-term interactions with the geologic environment. This activity will provide updated radionuclide solubility values, ranges, and uncertainties based on current understanding.

Analysis Plan 6: Engineered Barrier System Transport/Release

The objective of this plan is to define scenarios and pathways for radionuclide transport from the waste forms to the host rock, consistent with drift-scale water contact scenarios.

6.8.7 Thermohydrology Abstraction-Testing Workshop Results

Criteria for Prioritization

- Waste package temperature
- Relative humidity around the waste package
- Liquid water flow rate into the drift environment and onto a waste package
- Aqueous flow from the repository to the saturated zone.

Top Priority Issues for Thermohydrology

1. Issues related to thermohydrologic processes and parameters:

   - What model should be used for fracture–matrix interactions in total system performance assessment?

- How to upscale fracture properties and thermohydrologic processes.

- Should lateral (intra-unit) property heterogeneity be included in total system performance assessment?

2. Issues related to mountain-scale models:

   - What alternatives for repository design should be considered in mountain-scale models by total system performance assessment?

   - How important is the tradeoff between one-dimensional/two-dimensional modeling and three-dimensional modeling

   - How important is dual permeability at the mountain scale?

3. Issues related to drift-scale models:

   - Will variability of heat output among waste packages allow for condensate shedding onto cooler packages?

   - How to model seepage onto drifts and waste packages under non-isothermal conditions.

   - Is it necessary for total system performance assessment to provide drift-scale models that represent repository edge as well as repository center conditions?

4. Issues related to coupled processes:

   - Will phase-change processes cause chemical deposition and thus alteration to fracture and matrix properties?

   - Will thermal stresses cause significant hydrologic-property alterations in regions of compression and tension?

   - What effects would drift collapse have on temperature of the waste package, relative humidity in the drift, and seepage water contacting a waste package?

<u>Synopsis of Analysis Plans for Addressing Top-Ranked Thermohydrology Issues</u>

Analysis Plan 1   Mountain-Scale Thermohydrologic Abstraction and Testing

The objective of this plan is to provide abstraction information at the scale of the mountain. This will include development of thermally altered flow fields (both gas and liquid) both above and below the potential repository. Temperature and liquid saturation fields will be determined for the mountain and specifically at important strata such as the Paintbrush nonwelded tuff hydrogeologic unit (PTn), basal vitrophyre, zeolites, and the potential repository horizon. The effects of model dimensionality will be assessed using the three-dimensional site-scale thermal model and existing two-dimensional models currently in use. The same base case model will be used f both unsaturated zone flow and unsaturated zone transport analyses to ensure consistency in the final representation of the mountain.

Analysis Plan 2: Abstracting Drift-Scale Temperature, Relative Humidity, Liquid Saturation, and Liquid-Phase Flux as a Function of Location in the Repository

The objective of this plan is to provide abstraction information in the drift environment and at the waste package itself. This will include predictions of temperature and relative humidity at the waste package surface. Waste form temperatures will be computed using existing waste package models and the other results from this task. Drift wall temperatures and liquid rock and invert saturations will also be calculated. The analyses will use existing drift-scale models for different waste package types. Different repository locations (e.g., center and edge) will be captured by nesting drift-scale model domains into mountain-scale models.

Analysis Plan 3: Thermal-Hydrologic Modeling of Seepage into Drifts

The objective of this plan is to provide additional abstracted information for the drift environment. This task will focus on liquid water seepage into the drift and onto "hot" waste packages. This model is at the scale of the drift and will apply alternative conceptual flow models. Investigations will include dual permeability model and a "Weeps" flow model modified to include evaporation processes.

Analysis Plan 4: Coupled Processes Abstraction and Testing Plan

The objective of this plan is to perform sensitivity studies of the coupled processes at the mountain and drift scales. Analysts will determine if it is necessary to include flow property changes resulting from chemical and/or mechanical processes in the flow predictions for the total system performance assessment calculations.

6.8.8 Near-Field Environment Abstraction-Testing Workshop Results

Criteria for Prioritization

- Effect on dissolved radionuclide concentration
- Effect on colloidal radionuclide abundances
- Effect on in-drift sorption capacities
- Effect on in-drift porosity and permeability.

Top Priority Issues for Near-Field Environment

1. Issues related to solid phases:

    - Volume and flux of water in drift

- Compositions, abundances, and distribution (cement, alloys, organics, microbes, ceramics)
- Aqueous and gas reactions on materials
- Aqueous and gas reactions (corrosion) on waste packages
- In-drift system open or closed.

2. Issues related to gas phase:
   - Gas flux
   - Reactions with solids and microbes (excluding waste package)
   - Reactions with waste package
   - Thermal effects (water reactions)
   - Temporal heterogeneity.
3. Issues related to the aqueous phase:
   - Aqueous phase reactions with major introduced materials (excluding waste package)
   - Open versus closed system
   - Aqueous phase reactions with waste package
   - Temporal evolution of aqueous phase composition.
4. Issues related to colloids:
   - Reversibility of radionuclide sorption onto colloids
   - Water-composition effects
   - Waste form.

<u>Synopsis of Analysis Plans for Addressing Top-Ranked Near-Field Environment Issues</u>

Analysis Plan 1: Near-Field Environment Water-Solid Chemistry Model

The main objective of this plan is to develop a model of the water compositions that (a) are likely to react with the waste package and the waste form and (b) form the medium for transport through the engineered barrier system. The primary products of this effort are expected to be time-dependent bounds on the ranges of dissolved constituents needed as inputs to subsystem models like waste package corrosion (e.g., pH, chloride, fluoride, silica, carbonate, sulfate, calcium, and sodium). Analyses will be conducted over a number of scenarios, including variable starting water and gas compositions and extent of equilibration for reaction with concrete, backfill (crushed tuff), and steel sets. The results of the analyses will be cast as either ranges of concentrations or response hyper volumes.

Analysis Plan 2: Sensitivity Study and Potential Abstraction for Colloid-Facilitated Radionuclide Transport through the Near Field and the Unsaturated Zone

For the colloid issues, the existing plan from the unsaturated zone transport workshop was augmented to address additional considerations of introduced colloids in the near field. The main objective of this combined plan is to conduct sensitivity studies to assess the contribution to radionuclide release from colloids and, if shown to be substantial compared with dissolved and gaseous radionuclide releases, provide an abstracted model for total system performance assessment. A simplified model will be developed to examine the effect on plutonium release from intrinsic plutonium colloids (plutonium hydrous oxide polymers) and plutonium sorbed on nonradioactive colloids (specifically iron oxides). These analyses will consider interaction of plutonium with the solids in the drift and the rock minerals in both the rock matrix and fracture system.

Analysis Plan 3: Abstraction of the Effects of Microbial Communities on the Near-Field Geochemical Environment

The main objective of this plan is to develop response surfaces or analytic approximations that bound the effects of microbial processes on pH and gas composition evolution mainly carbon dioxide (CO_2) after emplacement. The three products of this plan will be (1) a spatial-temporal description of pH, (2) the temporal generation of CO_2, and (3) the temporal evolution of microbial population (i.e., biomass) as affected by nutrient availability, relative humidity, temperature, microbial reaction rates, and initial microbial community. These outputs will be cast in the form of response surfaces for the above parameters and fed to subsystem performance models, as well as to the other planned activities resulting from this workshop.

Analysis Plan 4: Abstracted Evolution of Gas Composition Throughout the Repository Drifts

This plan will address potential temporal changes in four gas constituents (water vapor, carbon dioxide, oxygen, and nitrogen). The main objective is to assess the competition between external drivers (i.e., the flux into the drifts) and in-drift source/sink terms for these gas constituents. The results will be given as response surfaces (with uncertainties) for the oxygen and CO_2 content of the gas in the drift through time. Initially, simple mass balance calculations between incoming gas and the capacity of source/sink terms to affect that incoming composition will be performed to assess the need for calculating more complex interactions. The analyses will consider a range of possible system permeabilities and two locations within the potential repository (center and edge) to evaluate sensitivities in the system. The response surfaces will constitute direct inputs to the waste package and waste form subsystems, and would be used by other abstraction-testing activities resulting from the near-field geochemical environment workshop.

6.8.9 Interactions with the NRC

Performance assessment staff participated in the DOE/NRC Technical Exchange on the Probabilistic Volcanic Hazard Analysis held February 25 26, 1997, in White Flint, Maryland. The performance assessment presentation discussed how the probability-distribution estimates derived from the probabilistic volcanic hazard analysis experts will be used in conjunction with the performance assessment consequence models to provide estimates of volcanic risk for the total system performance assessment for the viability assessment. In more detail, the talk reviewed the Total System Performance Assessment - 1991 models (direct releases at the surface) (Barnard et al., 1992; Eslinger et al., 1993) and the Total System Performance Assessment - 1993 models (indirect effects of volcanism on the ground-water transport source term) (Andrews et al., 1994; Wilson et al., 1994). The implications drawn from prior work were that Total System Performance Assessment - 1991 used very conservative models for waste entrainment and Total System Performance Assessment - 1993 showed insignificant radiation doses from the indirect effects of a dike intrusion.

A preliminary strategy for the total system performance assessment for the viability assessment includes adapting new work on volcanic entrainment, dike plumbing, and dissolution into the Total System Performance Assessment - 1991 model to make it more realistic. Indirect effects will also be incorporated by using new work on heat transfer and gas flow from nearby intrusive dikes. These indirect effects will be considered for all dike-waste interactions regardless of whether direct entrainment is modeled. Additionally, the alteration of ground-water flow patterns at the potential repository site from a nearby dike may also be modeled.

Forecast: The remaining two workshops will be held: (1) saturated zone ground-water flow and radionuclide transport and (2) biosphere (which includes radiation doses). The plans developed in all the nine workshops will be implemented.

6.9 CONTAINMENT BY WASTE PACKAGE (SCP SECTION 8.3.5.9)

SCP Section 8.3.5.9 addresses Issue 1.4, which asks whether the waste package will meet the performance objective for containment as required by 10 CFR 60.113.

Waste package container designs, as described in the Controlled Design Assumptions Document (CRWMS M&O, 1996c), currently focus on a multibarrier approach and include families of materials other than the copper-base materials and the iron to nickel-base "austenitic" materials that were the subject of the SCP conceptual design (SNL, 1987). The only option of these "alternate materials" currently being pursued is the "bimetallic/single metal," which is the multibarrier design in the advanced conceptual design report (CRWMS M&O, 1996b). Thus, progress on evaluating these "alternate materials" is discussed under Performance Assessment Activity 1.4.2.4 (Section 6.9.6 of this progress report) and Performance Assessment Activity 1.4.3.3 (Section 6.9.9 of this progress report).

6.9.1 Performance Assessment Activity 1.4.1.1 - Integrate Design and Materials Information (Metal Container)

The current waste package container designs focus on a multibarrier approach for both spent nuclear fuel packages and for vitrified high-level waste packages. These designs are robust in the sense that (a) multibarriers provide reinforcement to the containment function, (b) thick sections are used for some of the barrier materials, and (c) some of the barrier materials are very highly resistant to corrosion under a wide range of environmental conditions.

Engineered Materials Characterization Report

Revision 1 of Volume 3 on of the Engineered Materials Characterization Report (McCright, in prep) has been submitted to YMSCO for review and approval. Volume 3 contains the results of testing and modeling activities that have occurred since the report was originally issued in December 1994 as Revision 0 (Van Konynenburg and McCright, 1995). Volume 1 (on the background and history of the engineered barrier system candidate materials) and Volume 2 (on the physical and mechanical properties of the candidate materials) were not revised. Current corrosion test data and model development are discussed in Sections 6.9.6 and 6.9.9 of this progress report.

Forecast: Following YMSCO approval, Revision 1 of Volume 3 of the Engineered Materials Characterization Report will be published.

Waste Package Materials/Design Interface Activities

Integration between the waste package development effort and the waste package materials effort has continued with technical discussions and exchanges of weekly and monthly progress reports. Integration meetings are regularly scheduled between the two areas. The general fabrication techniques for the multibarrier container and how the barriers will be configured have been discussed between the two areas. In particular, fabrication techniques and barrier configuration impact the galvanic interaction between the barriers and the extent galvanic protection provided by the outer barrier can prolong the containment life of the inner barrier. Contact between the metals is an important factor in determining the effectiveness of galvanic protection.

The waste package design group is evaluating different processes for fabricating and welding the waste package containers. A "shrink-fit" arrangement is made by heating and expanding the outer barrier and slipping it over the inner barrier and then allowing the outer barrier to cool and contract. This ensures a reasonably intimate bond between the two metals over nearly all the surface area. The contact area between the two materials making up the galvanic couple is an important materials test parameter, so useful information will be obtained from and shared with the waste package design group to evaluate container fabrication processes.

Forecast: Waste package materials/design interface activities will continue because of the many technical issues common to both groups. In particular, discussions and information exchanges on the container fabrication/welding process evaluations will continue to receive attention because many aspects of the container performance are related to these processes.

Addition of Alloy 625

Since Progress Report #15, specimens of Alloy 625 (ASTM B 443), a nickel-chromium-molybdenum alloy, were purchased and added to the corrosion testing program (CRWMS M&O, 1996y), specifically the long-term comprehensive corrosion test, the electrochemically-based corrosion tests, microbiologically influenced corrosion tests, galvanic corrosion tests, and the humidity chamber oxidation/corrosion tests. These tests are discussed in Section 6.9.6 of this progress report.

Forecast: Alloy 625 is now fully integrated into the metallic barriers testing effort

Backfill Materials Study

No additional work has been performed on the effects of chemical additions to the backfill to buffer the pH and the Eh of the environment, since that reported in Progress Report #15 and incorporated into the draft Engineered Barrier System Performance Requirements Study Report (CRWMS M&O, 1996bb).

Forecast: No additional work is planned on backfill materials in the next reporting period.

Thermal and Chemical Degradation of Concrete and Invert Material

This is a new study that began in FY 1997. The objective of this study is to identify effects of the interaction between engineered barrier materials and water or water vapor in the potential repository. These engineered barrier system materials include, in addition to construction materials, introduced air and water, crushed tuff or muck rock as backfill or invert material, and introduced or enhanced microbial populations. In particular, the interest is in those effects that may be outside the bounds of predictions based on thermally perturbed rock. Complementary, but nonredundant studies of introduced materials are described in Section 5.2.6 of this progress report. The present experiments were intended to support a design decision regarding the use of precast concrete liners for mechanical support in repository emplacement drifts. In such a location the concrete will be subjected to elevated temperatures of at least 150 to 200°C and perhaps even higher if backfill is used.

An important objective of this work is to study the potential influences of microbial populations on an environment containing large volumes of concrete. A set of experiments has been developed to provide data for comparison with data from abiotic studies of introduced materials (see Section 5.2.6 of this progress report). The main objective was to understand the use of concrete by a microbial community as a function of provided macronutrients, carbon, nitrogen, phosphorus, and sulfur. The first matrix of 40 experiments evaluates microbial activity as a function of selected combinations of high, intermediate and starvation levels of the macronutrients at two different temperatures, 25 and 50°C, in a microcosm of crushed tuff. This was to provide the baseline information for the second set of experiments, which are identical with the exception of added crushed ESF concrete invert. A further objective of this work has been to compile experimental data and to assess the relevance of the available thermodynamic data for the long-term chemical modeling of engineered barrier materials. The emphasis this year is on the corrosion products of metals used in construction in general and on iron-based alloys in particular.

During the first few months, an activity plan was written and approved, and the necessary staff was hired and trained. The required glassware, plumbing, incubators and sterile hoods have been purchased. Presently, the scoping experiments are being plumbed in order to test filters, flow rates, and sterile controls. Sterile controls for this experiment must be extremely well characterized, because they must be chemically indistinguishable from the actual tests. The chemical signatures (cations, anions and total organic carbon) of five different sterilization methods are currently being tested in order to choose the most appropriate method. The chosen method will either have an obvious signature in a range that is not significant to the experiments (e.g., mercury), or will have an insignificant signal that is within the experimental error of the experiments.

With respect to the data collection and modeling studies, data from corrosion experiments conducted on potential waste package materials have been examined. The present GEMBOCHS thermodynamic data base may not be adequate to simulate these corrosion tests.

Forecast: Scoping experiments will establish an experimental protocol and should be completed soon. The first matrix of experiments are expected to be running in the next two months, and the first chemical results will be available very soon thereafter. The duration of the experiment is determined by the amount of time required to achieve chemical stasis. This will be determined to some extent during the scoping experiments but is expected to vary from experiment to experiment, depending on the chemical and thermal conditions. Ultimately, this information will be used to assess the ability to model long-term chemistry in the presence of microbial activity.

With respect to chemical simulation studies, the next effort will be to attempt to simulate some of the corrosion experiments (those conducted at 100 percent relative humidity and the long-term exposure aqueous experiments, discussed in Section 6.9.6 of this progress report), and thereby tailor the currently available kinetic parameters.

Performance Confirmation Plan Input

Input was prepared for the Performance Confirmation Plan (CRWMS M&O, in prep.[h]). The input covered the test and analysis scope sheets for the five testing concepts for the waste package, which were initially developed for the Performance Confirmation Concepts Study Report (CRWMS M&O, 1996z) in the previous reporting period and described in Progress Report #15. These five concepts are: (1) conduct laboratory measurements performed "off site" (meaning away from the repository); (2) perform in situ monitoring; (3) retrieve and characterize radioactive waste packages; (4) retrieve and characterize dummy waste packages; and (5) pull test specimens from various locations in the potential repository and characterize them. The characterizations would be performed at the potential repository or at an offsite location, depending on the complexity of the analysis and the facilities that will be constructed at the potential repository location. The test and analysis scope sheets included the objectives and descriptions of the concepts, the particular constraints inherent in each concept, and the requirements for facility construction, hardware, software, and data acquisition.

Forecast: As needed, the waste package materials organization will support systems engineering in the formal review and modification of the Performance Confirmation Plan.

6.9.2 Performance Assessment Activity 1.4.1.2 - Integrate Design and Materials Information (Alternate Barriers Investigation)

The purpose of this task is to characterize the behavior of ceramic materials and to determine degradation rates and mechanisms. One of the barriers incorporated in the disposal container may be fabricated from an oxide ceramic due to superior long-term aqueous corrosion performance. This activity is directed toward determining the feasibility of making a ceramic barrier part of the waste packages.

Survey

No survey work was done during this reporting period to evaluate alternative barrier designs, materials and processes in order to determine the feasibility of fabricating a satisfactory waste package.

Ceramic Coatings

In this reporting period, efforts were begun to both produce and evaluate ceramic coatings for carbon steel applied by various thermal spray techniques. An impervious oxide coating will protect a metallic substrate from contact with water and therefore corrosion. In general, thermal spray is a process of projecting molten droplets of metallic or ceramic materials onto a relatively cool surface so that they spread, cling, and solidify on impact. Many such droplets overlapped together form continuous coatings. The melting operation can be accomplished using an electric arc, by combustion or by detonation. Almost any surface can be coated, as long as the relative thermal expansions of the substrate and coating are well matched and as long as the surface is suitably roughened in advance (as by grit blasting)

Various thermal spray systems are being investigated in an attempt to evaluate which will produce the best coatings. The overall evaluation includes alumina, titania, combinations of these two materials and magnesium aluminate spinel. Of these, alumina is most desirable from a cost perspective. A contract was arranged with Vartech Inc. at Idaho Falls, Idaho, to generate samples of the listed materials using two variations on arc plasmas and a high-velocity oxy-fuel system. This work is ongoing.

For initial work, aluminum oxide was sprayed [illegible] a direct-current electric arc plasma. Die penetrant and metallographic studies began to characterize the resultant coatings and provide a working knowledge of the properties produced under various conditions. These procedures will be used in evaluations of coating materials and coating processes.

Thermal Transformation

Because two distinct structural forms of alumina can exist in a rapidly quenched coating, thermal transformation studies began to determine whether the transformation might take place over time at repository temperatures. The relative proportions of phases was determined for plasma-sprayed alumina samples using x-ray diffraction. These samples were then placed in separate furnaces held at temperatures of 300, 600 and 900°C. The steel substrates failed at 900°C, but the others are being held at their respective temperatures for periodic sampling. The first samples were withdrawn after six weeks, giving no clear indication that any transformation had taken place. The sampling process will continue for an indeterminate period unless the transformation is found to occur, allowing a lifetime prediction.

Impact Studies

An experimental matrix was designed to include impact testing on coatings using a 2-m drop tower to simulate rock fall in the repository. A ceramic impactor of appropriate chemistry and density will be used to represent the Yucca Mountain welded tuff. The rock itself is too variable on a small scale to provide reproducible results and machining costs to achieve the appropriate shapes are too high in any event. A slightly porous porcelain from which to manufacture the impactors was formulated to simulate welded tuff.

Corrosion Studies

In preparation for corrosion studies to follow, substrates were prepared consisting of cylindrical coupons 6 in. (150 mm) long and 1 in. (25 mm) diameter, with hemispherical ends. Suitable racks have been ordered to include these coupons in long-term corrosion studies (see Section 6.9.6 of this report).

Forecast: Studies begun in October 1996 will continue through the remainder of FY 1997. Thermal transformation specimens will be sampled at six months for further x-ray diffraction. The samples ordered from Vartech arrived in March 1997, which will allow microstructural and die penetrant evaluations to proceed. This, combined with additional samples, which may be obtained from other sources in the near term, will allow selection of both a material and specific coating process for long-term corrosion and impact testing.

6.9.3 Performance Assessment Activity 1.4.2.1 - Selection of the Container Material for the License Application Design

The objective of this activity is to select the container material for more detailed characterization of its properties relevant to attaining the performance objectives of the emplaced container. This activity involves the metallic materials and ceramic-metal systems, bimetallic/single metal systems, and coatings and filler systems.

Subactivity 1.4.2.1.1 - Establishment of selection criteria and their weighting factors The selection criteria and weighting factors for the viability assessment design were reported in Progress Report #15. These criteria and weighting factors remain in effect.

Subactivity 1.4.2.1.2 - Material selection The selection criteria and weighting factors were applied to the candidate materials, and reference materials were selected for the viability assessment design. That selection, which was discussed in Progress Report #15, remains in effect. The reference materials are listed in Section 5.1.4 of this progress report under Subactivities 1.10.2.4.2 through 1.10.2.4.7.

Forecast: No additional activity on disposal container material selection is planned for the viability assessment design phase, which includes all of FY 1997. Revision of the selection may be necessary, however, if the disposal container design is changed, if significant new information becomes available on the near-field environment or materials degradation, or if an even longer containment life is required of the waste package.

6.9.4 Performance Assessment Activity 1.4.2.2 - Degradation Modes Affecting Candidate Copper-Based Container Materials

The objective of this activity is to analyze which degradation modes have any significant chance of occurring on the candidate copper-based materials in the postemplacement periods and to perform laboratory testing and analysis activities to provide information for modeling the rate of degradation of the container materials (Performance Assessment Activities 1.4.3.1, 1.4.3.2, and 1.4.3.3, see Sections 6.9.7, 6.9.8, and 6.9.9 of this progress report).

Current designs, as described in the Controlled Design Assumptions Document (CRWMS M&O, 1996c), focus entirely on multibarrier waste package container configurations; therefore, degradation mode activities are reported under Performance Assessment Activity 1.4.2.4 in Section 6.9.6 of this progress report. The only option currently being pursued under the controlled design assumptions is the "bimetallic/single metal," which is the multibarrier design. See the beginning of Section 6.9 of this progress report.

No progress was made during this reporting period on the eight subactivities within this activity that address the degradation of copper-based materials and related laboratory testing and analysis; these were unfunded activities.

Forecast: See the forecast for Subactivity 1.4.2.4.3 in Section 6.9.6 of this progress report.

6.9.5 Performance Assessment Activity 1.4.2.3 - Degradation Modes Affecting Candidate Austenitic Materials

The objective of this activity is to determine which degradation modes have a significant chance of occurring for the candidate austenitic materials in the postemplacement periods and to perform laboratory testing and analysis activities to provide information for modeling the rate of degradation of the container materials (Performance Assessment Activities 1.4.3.1, 1.4.3.2, and 1.4.3.3, see Sections 6.9.7, 6.9.8, and 6.9.9 of this progress report).

Current designs, as described in the Controlled Design Assumptions Document (CRWMS M&O, 1996c), focus entirely on multibarrier waste package container configurations; therefore, degradation mode activities are reported under Performance Assessment Activity 1.4.2.4 in Section 6.9.6 of this progress report. The only option currently being pursued under the controlled design assumptions is the "bimetallic/single metal," which is the multibarrier design.

No progress was made during this reporting period on the nine subactivities within this activity; these were unfunded activities. The single exception is the inclusion of stainless steels in the test matrix for stress corrosion cracking (see Section 6.9.6 of this progress report).

Forecast: See the forecast for Subactivity 1.4.2.4.3 in Section 6.9.6 of this progress report.

6.9.6 Performance Assessment Activity 1.4.2.4 - Degradation Modes Affecting Ceramic-Metal, Bimetallic/Single Metal, or Coatings and Filler Systems

The objective of this activity is to evaluate potential degradation modes that can affect an alternative waste package container developed under the alternate barrier investigations and to perform the testing needed to quantify and model these degradation phenomena. These degradation phenomena apply to the post-emplacement periods. Laboratory testing and analysis activities are to provide information for modeling the rate of degradation of the container materials (Performance Assessment Activities 1.4.3.1, 1.4.3.2, and 1.4.3.3; see Sections 6.9.7, 6.9.8, and 6.9.9 of this progress report).

All the work discussed in the following subactivities is applicable to the bimetallic/single metal case for design alternatives discussed in the SCP (DOE, 1988).

Subactivity 1.4.2.4.1 - Assessment of degradation modes affecting ceramic-metal systems. Please refer to Section 6.9.2 of this progress report for discussion on the evaluation of ceramic coatings on steel substrates.

Subactivity 1.4.2.4.2 - Laboratory test plan for ceramic-metal systems of the alternate barriers investigations. Please refer to Section 6.9.2 of this progress report for discussion on the evaluation of ceramic coatings on steel substrates.

Subactivity 1.4.2.4.3 - Assessment of degradation modes affecting bimetallic/single metal systems. The objectives of degradation mode surveys are (a) to compile relevant previously published information about a candidate material and its performance in a number of environments and (b) to interpret this body of information in the context of a potential repository in Yucca Mountain. In many instances, the degradation mode survey indicates the ways in which a material can degrade and serves to indicate the rate and kind of degradation in environments that have some similarity to what a metal barrier may experience in the Yucca Mountain setting. In other instances, the lack of information suggests what work will be required to determine the behavior of the candidate material under Yucca Mountain environmental conditions.

The two most recently produced degradation mode surveys (Roy et al., 1996a; Goldberg and Dalder et al., 1995), discussed in Progress Report #15 have furnished the basis for some of the work discussed in Section 6.9.6 of this progress report.

Forecast: No additional surveys are planned for the immediate future, but welding effects in the corrosion allowance materials should be surveyed. The welding effects relative to this class of materials appear to be less of a performance is~ue than they are for corrosion resistant materials.

Subactivity 1.4.2.4.4 - Laboratory test plan for bimetallic/single metal material system.

L. g-Term Corrosion Studies

The objective of the long-term corrosion studies is to determine comprehensive corrosion properties of metallic alloys being considered for constructing the multibarrier waste package container. Three classes of materials are to be addressed: corrosion resistant, corrosion allowance, and intermediate corrosion resistant. Corrosion properties to be addressed are general corrosion, pitting corrosion, crevice corrosion, intergranular corrosion, stress corrosion cracking, hydrogen embrittlement, and galvanic corrosion. This activity will provide kinetic and mechanistic information about the corrosion degradation of candidate materials. This information will support materials selection, performance analysis, and model development. Tests are conducted in environments that bound the range of environmental conditions and water chemistries that are projected to develop near the container surface over long periods of time. These comprehensive corrosion tests are planned to last at least five years, with test specimens periodically removed and inspected to measure corrosion degradation as a function of exposure time.

Testing has begun in eight of the first twelve test vessels. The corrosion-allowance materials, carbon and low alloy steels, were emplaced in the first four test vessels, which contained dilute and concentrated aqueous solutions of near neutral pH at 60 and 90°C. The intermediate corrosion-resistant materials, 70/30 copper nickel and Monel 400, were emplaced in the next two vessels, which contained concentrated acidic solutions (pH 2.6) at 60 and 90°C. The corrosion-resistant materials, the nickel-chromium-molybdenum and titanium alloys, were emplaced in the next two vessels, which also contained concentrated acidic solutions (pH 2.6) at 60 and 90°C. The corrosion-resistant materials will be emplaced in the remaining four vessels, which will contain dilute and concentrated aqueous solutions of near neutral pH.

Alloy 625 test specimens were included in the corrosion-resistant materials testing. This material was added to the candidate list of corrosion resistant materials at the request of the waste package design group. A full compliment of specimens was purchased and characterized (weighed and measured). These specimens were emplaced in the test vessel with the other corrosion-resistant materials.

The first set of the corrosion-allowance specimens were withdrawn from a dilute solution test vessel at the end of March 1997. This represents six months of time in the test solution. The specimens will be analyzed for corrosion degradation. This information will be shared with the performance assessment and the waste package design groups.

Twelve additional test vessels will be installed and operational in FY 1997. The infrastructure for these additional test vessels has been installed. This includes the steel support structure, water and air feed lines, power feed lines, and the support structure for the electronic controls of the test vessels. The electronic and mechanical hardware for these vessels has been purchased. This includes the 6 kW heaters, the level sensors, the thermocouples, the condensers, the air flow meters, the power control units for the heaters, water feed solenoids, and the electronic support structure.

The twelve additional test vessels have been ordered. The vessels will accommodate the galvanic specimens and specimen testing in the concentrated alkalized solutions. To accommodate the pH 12 test solutions, a different containment material for the test vessels is necessary. They will be lined with a Teflon™-like material with better mechanical properties.

Galvanic specimens have been designed and quotes for fabricating the specimens have been obtained. Bids are being requested for fabricating the te t racks for the galvanic specimens.

Forecast: Within the next six months the first sets of corrosion-allowance test specimens will be withdrawn from the first four test vessels. This will represent six months of testing. The specimens will be analyzed for corrosion degradation. The order for the galvanic specimens will be placed. Twelve additional test vessels will be installed and operational within the next six months. These vessels will accommodate the galvanic specimens and the alkaline test solutions. Test racks will be designed to accommodate test specimens of potential basket materials, ceramic coated materials, and possibly fuel clad material specimens. These specimen racks will be inserted into the access port #6 of the test vessels. Access ports #1 to #5 of each test vessel are reserved for testing of the candidate materials. Galvanic specimens will be purchased and characterized (weighed and measured).

Humid Air Corrosion and Oxidation Studies

The objectives of this work are to determine the conditions under which aqueous film corrosion processes occur after the emplacement of the waste package and to characterize the mechanistic processes occurring. The conditions of susceptibility to aqueous film corrosion are particularly significant for a potential repository in the unsaturated zone, because the extent of degradation of the candidate materials becomes much greater when aqueous film processes begin. The key parameters appear to be relative humidity, temperature, gaseous contaminants, surface contaminants (salts), and surface condition of the metal. Thermogravimetric analysis is a particularly sensitive technique that uses a microanalytical balance to measure very small changes in weight gain as a material reacts with the environment. In addition, long-term testing under constant temperature and relative humidity in environmental chambers will give complimentary information.

Humid air corrosion on salt-covered (NaCl) carbon steel (A516 Gr55) specimens was investigated to understand the mechanistic aspects of the degradation process. The following discussion pertains to corrosion occurring under relative humidities greater than 70 percent and at a temperature of 80°C. The duration of the testing is of the order of 14 days.

A salt-covered specimen initially corrodes very fast. With time, however, the salt is "consumed" by the oxidation process, and the corrosion rate eventually ceases. At longer times the oxide transforms to a more stable oxide and spalls off the vertical surfaces.

X-ray diffraction studies indicate that during the initial stages of corrosion numerous crystalline oxide species are present on the surface. After the corrosion has ceased, the x-ray diffraction patterns indicate that hematite (Fe_2O_3) is a major component of the oxides on the specimen surface. The reddish-brown color of the oxide is also consistent with it being hematite. The oxide is also very porous and nonadherent.

Note that carbon steel species were visually monitored during testing and no visible water was observed even up to relative humidities of 90 to 95 percent. This is in contrast to salt-covered Alloy 625, a corrosion-resistant material, on which visible water was observed at 85 percent relative humidity. Decreasing the relative humidity below 70 percent resulted in rapid evaporation of the visible water. No visible corrosion occurred on Alloy 625.

Long-term testing has begun in an environmental chamber under constant conditions, 80°C and 50 percent relative humidity. Specimens are weight-loss coupons, which are clean, salt covered, and sandwiched (metal to metal) to create crevices. Initial materials being tested are carbon steels, Alloy 625, and a dilute titanium alloy, TiGr12. Numerous specimens are being tested to allow periodic removal for kinetic and mechanistic characterization.

Forecast: There will be continued investigation of the corrosion susceptibility of carbon steel at lower temperatures and with absorbed salts more typical of those present at Yucca Mountain. Aqueous salt solutions corresponding to those used in long-term corrosion studies will be used to deposit salts on the thermogravimetric test specimens. Testing will also be performed with elevated carbon dioxide (CO_2) levels that are speculated to be possible in the atmosphere at Yucca Mountain. Limited testing of the other candidate corrosion-allowance material, a low alloy steel, will be performed under similar conditions to make comparisons with carbon steel. Additional test specimens will be added to the environmental chamber, and testing in a chamber with higher relative humidity is scheduled.

Stress Corrosion Crack Growth Tests

The objective of these tests is to evaluate the susceptibility of candidate corrosion-resistant metallic container materials to environmentally assisted cracking, including stress corrosion cracking and hydrogen embrittlement under metallurgical and environmental conditions relevant to the potential underground repository. These test data will then be used in developing predictive models.

Stress corrosion cracking tests using fatigue-precracked and wedge-loaded double cantilever beam specimens began in November 1996. Results obtained so far indicate that Alloy 825 became susceptible to stress corrosion cracking upon exposure to the test environment for 30 and 60 days. Specimens were tested in acidified 5 percent salt (NaCl) solutions (pH 2.7) maintained at 90°C. The initial stress intensity was high and ranged from 33 to 52 ksi in$^{1/2}$. The stress intensity generally decreases as the crack grows. This combination of test conditions is severe, but the intent of this first series of experiments was to discern significant differences in the behavior of the candidate materials. The observed cracking in Alloy 825 appears to follow an intergranular pattern.

Forecast: Stress corrosion cracking tests using double cantilever beam specimens are ongoing, involving Alloys 825, G-30 and C-4 in a similar environment for durations ranging between 1 and 8 months. Tested specimens are being evaluated to determine the final wedge load and crack extension for calculating the critical stress intensity (K_{ISCC}) for stress corrosion cracking. The final crack length of each specimen after testing is being evaluated by metallography, which will enable an estimation of crack growth rate as a function of K_{ISCC}. Future tests using a similar approach will involve other corrosion-resistant materials such as Alloys C-22 and 625, and Ti Grade-12.

Microbiologically Influenced Corrosion Studies

The objective of microbiologically influenced corrosion studies is to determine if corrosion is enhanced by the presence and propagation of microorganisms, particularly bacterial species. Metabolic products from these microorganisms can alter significantly the chemical environment, and this can occur on a localized level or over a wide area of the container surface. Different microorganisms attack different alloys because of the metabolic diversity of microbial physiology combined with the chemical specificity of the corrosion process. This study surveys microbiologically influenced corrosion effects on the candidate waste package materials. The study also seeks to determine the causative biochemical reactions occurring under varying environmental conditions and to establish boundary conditions for microbial activity and propagation.

Testing of carbon steel specimens in microbially inoculated test cells at room temperature has been completed. The results of these studies were reported in Progress Report #15; in summary, it was found that a combination of sulfate-reducing, iron-oxidizing, and slime-producing bacteria demonstrated rates of corrosion five times greater than that shown in sterile, abiotic control cells incubated under the same conditions. This same experimental protocol, using the same sets of bacteria, is now being followed to determine corrosion rates of carbon steel test coupons at 50°C.

Concurrent with the carbon steel testing, some modifications have been made to the system for determining the corrosion rates of more corrosion-resistant alloys. Generally, the goal has been to provide the most auspicious conditions for corrosion, in order to detect corrosion of resistant materials, enable accelerated testing, and determine the greatest microbiologically influenced corrosion rates. Specifically, a varied formulation of UE-25 J#13 well water (which is also being used for the long-term corrosion testing) is being used as the solvent for the R2 media (Reasoner and Goldreich, 1985); this version of simulated UE-25 J#13 well water is used at ten-fold the concentration found in the well. Because of the higher concentration of salts, this increased electrolyte concentration improves the conductance of the media to facilitate electrochemical monitoring of these more corrosion-resistant materials. The R2 media has also been supplemented with 0.5 percent glucose and 0.75 percent proteose peptone #3 (Difco). These added nutrients have been found to promote the production of acids and sulfides, respectively, results of these studies are outlined in the Engineered Materials Characterization Report (McCright, in prep.), which are both central to the microbiologically influenced corrosion process.

In addition, the bacteria used in these studies are inoculated directly onto the metal coupons before the coupon is added to the test cell; this method of inoculation better enables direct contact of the test bacteria on the coupon. Thus far, cells containing both Alloy 825 and Type 304 stainless steel (for comparison purposes) have been inoculated with acid-producing and slime-producing bacteria isolated from the Yucca Mountain site, alone and in combination. These test cells have been incubated at room temperature while preliminary polarization resistance measurements are conducted and compared with sterile, uninoculated corrosion cells. Thus far, after only two to three weeks of incubation, there has been no apparent evidence of corrosion of these alloys. However, both monitoring protocols (i.e., scan window and rate) and incubation conditions may have to be altered to detect corrosion of resistant metals using this method. For example, as stated previously, the semisolid state of the media caused by the inclusion of agar inhibits diffusion of oxygen into the system; alteration to a completely fluid incubation system may alleviate this limitation and increase corrosion rates.

Forecast: The 50°C tests on 1020 carbon steel exposed to a variety of microbial organisms will continue. Iron-oxidizing bacteria, enriched from Yucca Mountain tuff, and sulfate-reducing bacteria are now being grown from stock cultures to extend the analysis of their corrosion potential to more resistant candidate alloys. Both Alloys C22 and 625 will also be incorporated into the testing program; 1020 carbon steel will also be retested under the improved incubation conditions. It is also planned to expand the size of the coupons to facilitate an improved signal-to-noise ratio. Thus, acid-producing, slime-producing, iron-oxidizing, and sulfide-producing bacteria (all isolated from Yucca Mountain) will be inoculated into the improved testing system, and assayed for their possible effect on the corrosion of various inner barrier candidate materials.

Electrochemically Based Corrosion Studies

The purpose of this study is to evaluate the susceptibilities of candidate waste package container materials to localized corrosion, such as pitting and crevice corrosion, in a range of localized environments possible in the potential repository. Pitting is one of the most destructive and insidious forms of corrosion and requires an extended initiation period before visible pits appear. Pitting is an autocatalytic process, because the corrosion processes within a pit produce conditions that are both stimulating and necessary for the continuing activity of the pit. Crevice corrosion is usually associated with a small volume of stagnant solution caused by holes, gasket surfaces, and lap joints. This type of damage is believed to be the result of differences in metal ion or oxygen concentration between the crevice and its surroundings. This study focuses on determining critical potentials for the onset of pitting corrosion and crevice corrosion.

Electrochemical cyclic potentiodynamic polarization experiments involving iron-nickel-chromium-molybdenum alloys (Alloys 825, G-3 and G-30), nickel-chromium-molybdenum alloys (Alloys C-4, C-22 and 625), and titanium-base alloy (Ti Grade-12) are ongoing. Cyclic potentiodynamic polarization experiments involving all these alloys in brines of various salt content (1 to 10 wt% NaCl) and pHs (2-3, 6-7, and 10-11) at ambient and elevated temperatures (up to 90°C) were completed, the results of which were presented in two recent reports (Roy et al., 1996b and 1997).

Results indicate that Alloys 825, G-3 and G-30 underwent pitting and crevice corrosion in all tested environments, with Alloy 825 showing the maximum susceptibility (Roy et al., 1997). As to the localized corrosion behavior of nickel-chromium-molybdenum alloys, Alloy C-4 suffered from pitting in all tested environments. But the extent of pitting was less severe than that observed with iron-nickel-chromium-molybdenum alloys. Alloy C-22 and Ti Grade-12 were immune to localized attack under all experimental conditions tested.

Consistent with the results of other investigators, for alloys susceptible to pitting, the critical pitting potential (E_{pit}) in acidic brines was shifted to more active (negative) values with increasing chloride ion (Cl^-) concentration. The mechanism for transition from passivity to pitting in susceptible alloys may be based on reversible competitive adsorption of Cl^- into the oxide-liquid interface (double layer) with oxygen for sites on the alloy surface. At a sufficiently high concentration corresponding to E_{pit}, Cl^- ions succeed at favored sites in destroying passivity by displacing adsorbed oxygen ions. For Alloy C-22 and Ti Grade-12, which showed sufficiently noble critical potential to overlap the transpassive region, formation of protective oxide films on alloy surface resulting from oxygen evolution from electrolysis of test solutions may possibly account for the enhanced resistance to pitting corrosion.

In brines containing 10 weight percent NaCl, E_{pit} for susceptible alloys was shifted to more noble (positive) values because of a change in pH from acidic to neutral. At alkaline pH, Alloys G-3, G-30, and C-4 showed somewhat lower E_{pit} values compared with those in neutral brines. For Alloy 825, E_{pit} was shifted to a slightly more noble value in alkaline brine. The more active E_{pit} value for susceptible alloys in acidic brines may be the result of the acceleration of cathodic reaction caused by high concentration of hydrogen ions. The inhibitive effect of hydroxyl ions may possibly account for more noble E_{pit} value at alkaline pH.

Consistent with the results of other investigators, E_{pit} became more active with increasing temperature, suggesting the occurrence of a temperature-induced change in properties of protective surface films. As to the effect of electrochemical potential scan rate on E_{pit}, a general trend was not observed that would be valid for all alloy-environment combinations studied. E_{pit} response to scan rate appears to be a function of the kinetics of passive film formation at applied potentials.

No consistent pattern on the effect of Cl^- concentration, temperature, and pH on corrosion potential (E_{corr}) and protection potential (E_{prot}) was observed.

As to the corrosion behavior of Alloy 625, the results indicate that this material suffered from pitting, crevice, and intergranular corrosion in all tested environments under potentiodynamic control. Metallographic evaluation is ongoing on Alloys 625 and C-22 specimens (unused and tested) to study the microstructural characteristics of both these alloys, and to relate them to their performance in cyclic potentiodynamic polarization experiments. Preparation of a technical report based on these findings is in process.

Cyclic potentiodynamic polarization experiments involving all seven candidate inner container alloys were also performed at 60 and 90°C in dilute and concentrated aqueous environments containing species present in well UE-25 J#13 water. The pH of these solutions ranged from 8.50 to 9.00. More complex shapes of the polarization curves were observed than those obtained from simple chloride solutions. The electrochemical data obtained from this study are currently being evaluated.

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Cyclic potentiodynamic polarization experiments at 60°C and 90°C using several corrosion-allowance a. d corrosion-resistant alloys in aqueous solutions containing 6 wt% ferric chloride ($FeCl_3$) have l -gun. The significance of these experiments is to determine if there is a possible detrimental effect from corrosion products generated from the outer barrier material on the performance of the inner barrier. Particularly, if ferric corrosion products are formed under acidifying conditions (such as in a crevice or in the presence of acid-producing bacteria), then localized corrosion of the inner barrier may be enhanced despite otherwise favorable galvanic effects between the two metals.

Longer-term electrochemical polarization experiments at controlled potentials (potentiostatic) were initiated to evaluate the initiation and growth of stable pits in susceptible environments using Alloy 825. Current tests are being performed at ambient temperature using controlled potentials based on measured E_{corr} value in an acidic brine to establish the pit initiation and growth behavior as a function of the combination of materials and environment over a pre-set duration. The magnitude of the applied controlled potential will gradually be modified (made more noble with respect to E_{corr}) if no pits are initiated. A similar technique will be used for elevated temperature (up to 90°C) potentiostatic polarization tests.

Forecast: Cyclic potentiodynamic polarization experiments performed in 60 and 90°C iron chloride ($FeCl_3$) solution will continue. Potentiostatic polarization tests at various controlled electrochemical potentials both at ambient and elevated temperatures will also continue. Results from both types of tests will be used in model development (see Section 6.9.9 of this progress report).

Galvanic Corrosion Studies

The objective of these studies is to provide an understanding of the electrochemical interaction between the dissimilar metals proposed for multibarrier waste package designs. The corrosion-allowance outer barrier is expected to provide galvanic protection to the inner barrier. A first objective is to evaluate the effectiveness of this protection. A second objective is to determine if this protection will function in all circumstances. Even though the thicker outer corrosion-allowance barrier, under the current all-metallic multibarrier waste package design concepts, may provide corrosion protection to the inner corrosion-resistant barrier, galvanic corrosion resulting from the breaching of the outer container may impact the performance of the inner container. Therefore, the galvanic corrosion behavior of inner and outer container materials must be evaluated to predict their service lives. Galvanic corrosion can be defined as accelerated corrosion of a metal because of an electrical contact with a more noble metal while exposed to a common electrolyte.

A literature survey on galvanic corrosion behavior of different candidate corrosion-allowance and corrosion-resistant metallic container materials in various environments was presented in a recent report (Roy et al., 1996a). Galvanic corrosion susceptibility of welded joints containing dissimilar metals was also discussed in this report. Even though the precise environment surrounding the waste packages in the potential underground repository is unknown, the various tested environments cited in this report should cover possible environmental conditions that might be encountered by the candidate container materials.

Environmental factors (such as temperature, pH, and electrolytic composition) and metallurgical factors (such as surface condition and thermomechanical history) can influence the galvanic corrosion involving two or more dissimilar conducting materials. Apart from these parameters, factors such anode-to-cathode area ratio, distance between electrodes, and geometric shapes are unique to galvanic corrosion. Accordingly, galvanic corrosion tests taking all these factors into consideration and using a modified cell were initiated in January 1997. These preliminary experiments are currently being performed at ambient temperature using corrosion-allowance material (A 516) as an anode, and a corrosion-resistant alloy (either Alloy 825 or G-3) as a cathode, galvanically connected in an acidic brine by means of a potentiostat. This A 516 is compositionally similar to 1020 carbon steel used in other metallic barrier corrosion studies. An equal area of anode and cathode is being used in these tests. Eventually tests are being planned at two other area ratios and electrode distances. The data generated from these experiments are usually presented as either the current density versus time, or the potential versus time. These test data are currently being evaluated and will be reported later.

Forecast: Potentiodynamic polarization experiments involving A 516 steel are planned in various environments to superimpose the resulting polarization curves on the polarization curves of corrosion-resistant alloys, previously tested in similar environments, to estimate the equilibrium potential (mixed potential) and current density from galvanic coupling in a common electrolyte. These data will then be compared with those obtained from the zero-resistance ammeter currently being used. The current galvanic corrosion testing will be extended to other couples involving other corrosion-allowance and corrosion-resistant materials at ambient and elevated temperatures. Anode-to-cathode area ratio will be modified (greater than or less than one). Furthermore, the electrolytic resistance will be modified by varying the distance between the two electrodes.

6.9.7 Performance Assessment Activity 1.4.3.1 - Models for Copper and Copper Alloy Degradation

Current designs, as described in the Controlled Design Assumptions Document (CRWMS M&O, 1996c), focus entirely on multibarrier waste package container configurations; therefore, modeling activities are reported under Performance Assessment Activity 1.4.3.3 in Section 6.9.9 of this progress report.

No progress was made during this reporting period for the modeling of various degradation processes of copper and copper-alloy materials; these were unfunded activities.

Forecast: See the forecast for Performance Assessment Activity 1.4.3.3 in Section 6.9.9 of this progress report.

6.9.8 Performance Assessment Activity 1.4.3.2 - Models for Austenitic Material Degradation

Current designs, as described in the Controlled Design Assumptions Document, focus entirely on multibarrier waste package container configurations; therefore, modeling activities are reported under Performance Assessment Activity 1.4.3.3 in Section 6.9.9 of this progress report.

No progress was made during this reporting period on this activity and the eight subactivities for modeling of various degradation processes of austenitic materials; these were unfunded activities.

Forecast: See the forecast for Performance Assessment Activity 1.4.3.3 in Section 6.9.9 of this progress report.

6.9.9 Performance Assessment Activity 1.4.3.3 - Models for Degradation of Ceramic-Metal, Bimetallic/Single Metal, and Coatings and Filler Alternative Systems

The modeling work discussed below applies to the bimetallic/single metal design alternative, as discussed in the beginning of Section 6.9 of this progress report.

Subactivity 1.4.3.3.1 - Models for ceramic-metal systems. Work in this subactivity is just starting. It is being coordinated with the progress made in demonstrating the feasibility of applying a thermally sprayed ceramic to a metal substrate.

Subactivity 1.4.3.3.2 - Models for degradation of bimetallic/single metal systems.

Performance Assessment Model Abstractions

Work reported in Section 6.8.4.

Oxidation and Corrosion Model for the Outer Barrier Material

Work continues on a deterministic model to predict long-term effects of low-temperature oxidation. The material of focus is carbon steel, the principal candidate for the outer barrier container material. Initial emphasis is on humid-air oxidation, in which atmospheric water influences oxidation through condensing on hygroscopic surface contamination or as thin films. Once aqueous conditions are established on the carbon steel surface, the degradation rate becomes higher, and the pH and several other chemical parameters influence the degradation rate. This work will eventually be joined with a companion model on general aqueous corrosion—meaning more-or-less uniform (as opposed to pitting) corrosion in the presence of bulk water.

The transitions between the three regimes of dry oxidation, humid-air oxidation, and general aqueous corrosion will be modeled according to experimental studies of the critical relative humidity for the onset of aqueous effects on the metal surface. The experimental work is discussed in Section 6.9.6 of this progress report. As previously reported, physically based model calculations lead to the conclusion that dry oxidation will not significantly degrade the performance of thick, corrosion-allowance materials for hundreds to thousands of years. These model calculations were recently presented (Henshall, 1996). Modeling of humid-air corrosion is in its early stages. Most of the work has involved reviewing relevant theories and experimental literature.

Forecast: Further studies are in progress to consider internal oxidation (particularly important for the alloy steels). A deterministic model is planned for aqueous corrosion of the outer barrier and the progression of corrosion from dry to humid to wet conditions. The humid and wet conditions are expected to have terms expressing the rate as a function of temperature, pH, dissolved oxygen content, and chemical speciation in the atmosphere and in the water.

Inner Barrier Corrosion Model

The overall objective of this activity is to derive predictive tools that will enable performance assessment of candidate materials during extended periods of time while exposed to Yucca Mountain conditions. In particular, the inner barrier material may be subject to localized corrosion once the outer barrier is breached. Much of the modeling effort depends on the characteristics of the environment that eventually contacts the inner barrier surface. Many of the environmental characteristics must be projected from assumptions for different scenarios of how water will enter the repository drifts and contact the waste packages.

As a result of the January model abstraction workshop on waste package degradation, some rethinking occurred on how best to put together the various models for corrosion of the inner barrier. Most recently, attention has been directed towards developing a corrosion model to predict the rate of penetration of the corrosion-resistant inner barrier, a function of the near-field environment. Note that the near-field environment is characterized by temperature, humidity, in-drift water dripping, and the chemistry of the contacting water. There are several modes of generalized and localized corrosion that may play an important role in the ultimate failure of engineered barriers used for the geological disposal of high-level radioactive wastes. Penetration of the corrosion resistant material will be assumed to be from localized corrosion: pitting corrosion, active crevice corrosion, or both. This model will account for the interaction between the corrosion-allowance outer barrier and the inner barrier. Interactions will include pH suppression in the crevice caused by the hydrolysis of products from corrosion-allowance material corrosion, establishment of a protective mixed potential at the corrosion-resistant material surface, and eventual crevice corrosion of the corrosion-resistant material beneath the accumulated corrosion product. Several of these effects will be accounted for with a near-field environment correction (calculation of pH and mixed potential) applied at the interface between the corrosion-allowance material and corrosion-resistant material, before applying the stochastic pitting model, or before applying a general corrosion model.

At the waste package degradation abstraction-testing workshop, the following three hypotheses were formulated: (1) penetration of the outer barrier comprised of corrosion-allowance material will be by either humid air corrosion or aqueous corrosion; (2) the inner barrier composed of corrosion-resistant material will be exposed in patches as the outer barrier corrodes; and (3) the crevice region surrounding each exposed patch can be subdivided into three generic zones. These zones are defined as follows: Zone 1 is the corrosion-resistant material that will be directly exposed to the near-field environment, via humid air or a thin layer of oxygenated and acidified water; Zone 2 is the corrosion-resistant material that will be exposed to a thin layer of acidified water, with a gradient in oxygen concentration; and Zone 3 is the corrosion-resistant material that will be exposed to a thin layer of acidified and deoxygenated water. The corrosion phenomena in these zones may progress through two distinct phases. Furthermore, corrosion within the crevice progresses through two phases: Phase 1 is the active corrosion of the corrosion-allowance material crevice wall; and Phase 2 is the classical crevice corrosion of the corrosion-resistant material, with passive-active transition.

During Phase 1, the corrosion-allowance material wall will undergo active anodic dissolution, while the corrosion-resistant material wall will be maintained below the damage threshold. Initially, the crevice between the corrosion-allowance material and corrosion-resistant material will be filled with water as the temperature of the container drops below 100°C. This water will be acidified by the hydrolysis of corrosion products from the anodic dissolution of the outer barrier. The pH of this electrolyte is suppressed by various hydrolysis reactions involving dissolved iron. The corrosion potential of the corrosion-allowance material at the point of penetration will be located between the reduction-oxidation potentials for cathodic oxygen reduction and anodic iron dissolution, as well as hydrogen evolution. This value can be calculated from mixed potential theory, is expected to be below the damage threshold of the corrosion-resistant material, and should galvanically protect the corrosion-resistant material. Passivation of the crevice wall formed by the corrosion-allowance material will not be possible at the pH and potential maintained at the mouth of the crevice. The rate of formation of precipitated products from corrosion of the corrosion-allowance material wall will be greatest near the mouth of the crevice. These hydroxides and oxyhydroxides will accumulate near the crevice mouth, eventually filling the space.

During Phase 2, a second crevice will form between the precipitated, tightly packed corrosion products and the corrosion-resistant material. Initially, the two walls of this new crevice will be formed by the precipitated solids, which will be relatively dielectric, and the passive surface of the corrosion-resistant material. At a critical distance into this crevice (d_c), which can be calculated from classical corrosion theory, the potential of the corrosion-resistant material will pass through the passive-active transition (E_{pass}). At distances less than d_c, the localized chloride concentration, pH, and potential may possibly cause localized breakdown of the passive film (initiation of pitting and stress corrosion cracking). At distances greater than d_c, the corrosion-resistant material will experience active crevice corrosion. Note that classical crevice corrosion theory requires passive crevice walls near the mouth for the generation of a potential drop in the crevice. This potential drop causes depassivation within the crevice, at a critical distance from the mouth. This wall will begin to corrode at a rate limited by the availability of cathodic reactants necessary for depolarization of the anodic dissolution reaction. These reactants will probably be dissolved oxygen entering the crevice through the mouth, or hydrogen from the electrolyte.

Forecast: A first-order approach to the interaction between corrosion-allowance material and corrosion-resistant material will be to account for the effects of the corrosion-allowance material corrosion on the localized environment that exists in the crevice separating the outer and inner barriers. This will involve (a) a decrease in the pH caused by hydrolysis of corrosion products; (b) initial establishment of the mixed electrode potential below the damage threshold of the corrosion-resistant material; and (c) crevice corrosion attack of the corrosion resistant material beneath precipitated corrosion products. These interfacial effects will "set the stage" for localized attack (pitting, active crevice corrosion, etc.) of the corrosion-resistant material, and will be accounted for with an "interfacial near-field environment correction." This relatively simple correction will facilitate the application of other corrosion models at this boundary, including the stochastic pitting model described in Progress Report #15 (DOE, 1997e). Necessary data such as equilibrium constants for hydrolysis, oxygen solubility, diffusion coefficients, and electrokinetic rate constants can be gleaned from the literature and from the corrosion testing reported in Section 6.9.6 of this progress report.

Particular attention will be paid to the crevice that may form between the corrosion-allowance material and corrosion-resistant material, depending on how the container is fabricated and on the pattern of corrosion attack through the outer barrier. A deterministic crevice model will be used to define localized conditions within the crevice. This model will account for (a) pH shift to acidic values caused by the hydrolysis of dissolved iron, nickel, and chromium; (b) differential oxygenation in the crevice from mass transport limitations; and (c) the temporal and spatial dependence of general and localized corrosion within the crevice.

6.9.10 Performance Assessment Activity 1.4.4.1 - Estimate of the Rates and Mechanisms of Container Degradation in the Repository Environment for Anticipated and Unanticipated Processes and Events, and Calculation of Container Failure Rate as a Function of Time

Work on SCP Subactivities 1.4.4.1.1 (deterministic calculations) and 1.4.4.1.2 (probabilistic calculations) is described together because the work performed includes both deterministic and probabilistic aspects.

Performance Assessment Model Abstractions

Work is reported in Sections 6.8.4, 6.8.7, and 6.8.8 of this progress report.

Localized Corrosion Submodel

The localized corrosion submodel PIGS takes into account the possibility that pit growth may cease for deeper pits on a random basis. The model has been applied to a single data set so far. The model is able to reproduce the behavior of the actual data, lending credence to the postulated mechanisms. In this reporting period, a draft report was prepared (Henshall, in prep.) covering a description of the mathematical basis for the model and its application to one data set. Additional laboratory data have been obtained to attempt to provide additional confirmation of the model using experimental data.

Forecast: Enhanced subsystem models will be developed for intended use in model-testing sensitivity analyses and in the total system performance assessment for the viability assessment. The development will be guided by plans being developed in and subsequent to the workshops mentioned above (see discussion in Section 6.9.9 of this progress report). See also the forecasts of Sections 6.10.8 and 6.10.10 of this progress report.

6.9.11 Performance Assessment Activity 1.4.5.1 - Determination of Whether the Substantially Complete Containment Requirement is Satisfied

The objective of this activity is to use waste package modeling results from Performance Assessment Activities 1.4.4.1, 1.5.4.1, and 1.5.4.2 (see Sections 6.9.10, 6.10.9, and 6.10.10 of this progress report) to predict waste package containment performance using the scenarios and models developed in Performance Assessment Activities 1.5.3.1 through 1.5.3.5 (Sections 6.10.4 through 6.10.8 of this progress report). The results of these calculations will then be compared with the interpretation of substantially complete containment to determine whether the performance objective has been met for all times during the containment period.

No progress was made during the reporting period; this was an unfunded activity. The current emphasis for the total system performance assessment for the viability assessment is on total system performance measures, rather than on subsystem performance measures such as substantially complete containment.

Forecast: Improved modeling of substantially complete containment will become possible from developments in Sections 6.9.9, 6.10.8 and 6.10.10 of this progress report, although the near-term emphasis of those activities will be on release-rate modeling to support the total system performance assessment for the viability assessment.

6.10 ENGINEERED BARRIER SYSTEM RELEASE RATES (SCP SECTION 8.3.5.10)

SCP Section 8.3.5.10 addresses Issue 1.5, which asks whether the waste package and repository engineered barrier systems will meet the performance objective for radionuclide release rates as required by 10 CFR 60.113.

6.10.1 Performance Assessment Activity 1.5.1.1 - Integrate Waste Form Data and Waste Package Design Data

The objective of this activity is to accumulate the waste form data and waste package design data from waste producers, fuel manufacturers, and other repository studies. No tests or analyses are performed in this activity.

Subactivity 1.5.1.1.1 - Integrate spent nuclear fuel information. The spent nuclear fuel waste form testing data and modeling needs are being addressed in support of the Controlled Design Assumptions Document (CRWMS M&O, 1996c). Activity plans for oxidation tests, dissolution/release tests, and modeling dissolution/release are being revised. The information on spent nuclear fuel release modeling was incorporated in Section 3.5 of the Waste Form Characteristics Report, version 1.2 (Stout and Leider, 1996). Progress during this reporting period is presented in Sections 6.10.2 and 6.10.6 of this progress report. A workshop on waste form alteration and radionuclide mobilization was held in February 1997. The highest ranked issues were dissolution/alteration rate, release rate, solubility limits, colloidal kinetics, and cladding degradation. High burnup spent fuel test samples were also identified as an issue.

Current planning is in progress to identify and procure cladding segments of a high burnup spent fuel from United States fuel vendors.

Subactivity 1.5.1.1.2 - Integrate glass waste form information. The glass waste form testing data and modeling ...s are being addressed in support of the Controlled Design Assumptions Document. Information was incorporated in Section 3.5 of the Waste Form Characteristics Report, version 1.2 (Stout and Leider, 1996). Progress during this reporting period is presented in Sections 6.10.3 and 6.10.7 of this progress report.

Subactivity 1.5.1.1.3 - Integrate waste package and repository design information. No progress was made during the reporting period; this was an unfunded activity.

Forecast: The Waste Form Characteristics Report will be updated for the spent nuclear fuel and glass waste forms. This report is being revised section by section, rather than waiting to revise the entire document.

6.10.2 Performance Assessment Activity 1.5.2.1 - Characterization of the Spent Nuclear Fuel Waste Form

The objective of this activity is to conduct tests that will provide data on the release rate of radionuclides from the spent nuclear fuel waste form.

Subactivity 1.5.2.1.1 - Dissolution and leaching of spent nuclear fuel. The objective of this subactivity is to generate spent nuclear fuel dissolution data for use in performance assessments and for direct use in licensing. As part of this task, saturated flow-through tests on spent nuclear fuel and uranium oxides are designed to measure dissolution rates of these materials and their dependence on several parameters such as solution pH, temperature, oxygen fugacity, flow rate, and solution anions, particularly carbonate species. The unirradiated uranium oxides represent new or zero burnup fuel. In addition, unsaturated drip tests are used to determine the rate of fuel alteration and release rates of different radionuclides under conditions of low-water-volume contact rates. Colloidal actinide species may form under these unsaturated conditions and be a major transport mode for radionuclides.

In the spent nuclear fuel flow-through tests, effort was focused in two different areas: (1) uranium oxide (UO_2) matrix flow-through dissolution rate tests on pressurized water reactor and boiling water reactors fuels at a variety of burnups and alkaline and acidic pHs; and (2) gap inventory tests for iodine-129. Four flow-through tests with ATM-105 fuel in a mini-matrix (2×10^{-2} M carbonate and 2×10^{-4} M carbonate, both at 25 and 75°C) were completed, and the tests were terminated. The final steady-state dissolution rates observed for these four tests were essentially unchanged from the interim values. Three new flow-through tests with ATM-103 grain-size powder specimens were started: one test with pH = 6 nitric acid at 25°C, one test with pH = 6 nitric acid at 75°C, and one test with pH = 4 nitric acid at 25°C. These tests complement two ongoing tests with pH = 6 nitric acid at 25°C that are being conducted with two different size ATM-103 fuel fragments (~1 mm and ~5 mm fragments) to investigate grain boundary leaching rates.

Grain-boundary inventory measurements of iodine-129 were conducted with four different ATM-105 and ATM-106 specimens from different rods with differing fission gas release percentages. However, the results are questionable and, thus at least some of the tests will be repeated.

In the unsaturated tests, two commercial spent nuclear fuels, ATM-103 and ATM-106, are being tested in three types of tests: high drip rate tests, low drip rate tests, and vapor tests. A surrogate water, EJ-13, was produced by reacting water from ground-water well UE-25 J#13 with Yucca Mountain tuff for approximately 80 days at 90°C. The spent nuclear fuel in the tests has undergone 54 months of reaction at 90°C by the end of March.

Scanning electron microscopy examination of a section from the ATM-103 spent nuclear fuel fragment, which was from the high drip rate test after 3.7 years of testing, indicated that reaction occurred primarily as a front through the grains, with limited reaction down the grain boundaries. The depth of reaction was a minimum of 20 μm, the diameter of a totally reacted grain. The transmission electron microscopy examination of another section from the same ATM-103 fragment indicated that (a) technetium, molybdenum, and ruthenium were being removed from epsilon-phase particles in reacted areas of the fuel grains; (b) 1 to 2 weight percent ruthenium and molybdenum were being incorporated into the uranium silicate alteration product present on the surface of the spent fuel; (c) small amounts (parts per million) of technetium were also incorporated into the uranium silicate alteration product; and (d) plutonium appeared to be concentrating on the fuel surface at areas adjacent to reacted grains. Epsilon-phase particles are particles containing the fission products molybdenum, palladium, rhodium, ruthenium, and technetium in a UO_2 matrix.

All seven unsaturated tests were sampled after 4.1 years of reaction. Scanning electron microscopy results for identification of the alteration products in the ATM-103 vapor test indicate that the major alteration product was dehydrated schoepite ($UO_3 \cdot 0.8\ H_2O$). The presence of large quantities of dehydrated schoepite, a highly soluble uranium product, shows the extent of the spent fuel reaction in a short reaction time, 4.1 years, under unsaturated conditions with only water vapor present. The iodine-129, strontium-90, and technetium-99 data for the low drip rate and vapor tests for the 3.1 year reaction interval were analyzed. An increase in the release fraction for technetium-99 was noted in both ATM-103 tests. The release fractions for these two tests were only 50 and 100 times less, respectively, than the release fraction, $\sim 5 \times 10^{-3}$, observed in the high drip rate test at the 3.1 year reaction interval. An increase in the release fraction for iodine-129 was noted for both vapor tests, as well as the ATM-103 low drip rate test. The iodine-129 release fractions, $\sim 10^{-2}$, were comparable in vapor, low drip rate and high drip rate tests for ATM-103. No increase in the release fraction was noted for strontium-90, which remained at 1×10^{-6}. The results for the ATM-106 low drip rate test, in which the fuel had been rinsed in EJ-13 water for less than 20 minutes, could not be directly compared with the other data. Examination of a clay colloid collected in a high drip rate test after 3.1 years of reaction indicated that both technetium-99 and molybdenum were incorporated into the clay.

Aliquots of the leachate and acid strip solutions from the seven unsaturated tests after 4.1 years of reaction were prepared and submitted for inductively coupled plasma-mass spectroscopy analysis. Results of these analyses will be presented in future reports. The inductively coupled plasma-mass spectroscopy data from the 3.7 year sampling of the two high drip rate tests were analyzed and are being incorporated into existing data tables. Samples of the fuel after 4.1 years of reaction under vapor and low drip rate conditions were microtomed for later transmission electronic microscopy examination.

Dynamic light scattering is being developed as a method to study colloids formed from the reaction of glass and spent fuel with ground water under potential repository conditions at Yucca Mountain. The data obtained by dynamic light scattering will include size classes and concentrations of colloids present in the solutions. The dynamic light scattering unit is operational and samples from ongoing, waste form, corrosion tests will examined as they become available.

Forecast: Flow-through dissolution measurements on spent nuclear fuel will continue. There are still major gaps in the data base of flow-through tests for fuels with different burnups, most notably as follows: (a) the remaining three extreme conditions should be tested for the highest burnup fuel currently available for testing, ATM-106; (b) there are no data for low burnup fuels between zero (UO_2) and 30 MWd/kgM (ATM-103) (fuel specimens with burnups of about 15 to 20 MWd/kgM are available for testing from the ends of fuel rods on hand, and the current test plan calls for testing these fuels in the near future); and (c) fuels with burnups up to about 70 MWd/kgM should be tested, but none are currently available.

The unsaturated tests will continue. Detailed analysis of the leachates from the spent nuclear fuel tests will continue to determine whether there is a decrease in the leach rate and a change in the composition, form, and quantity of the radionuclides. The alteration products on the spent nuclear fuel will continue to be identified. Attention will be paid to the type and depth of reaction within a spent nuclear fuel fragment. A detailed test plan has been approved and will be implemented to address issues identified during the ongoing testing of spent nuclear fuel. Additional spent nuclear fuel samples from the unsaturated tests will be cross-sectioned so that a three-dimensional representation of the sample reactions and a more quantitative determination of crystal phase compositions can be obtained.

<u>Subactivity 1.5.2.1.2 - Oxidation of spent nuclear fuel</u>. The oxidation of spent nuclear fuel under potential repository conditions depends primarily on temperature and time after the spent nuclear fuel is exposed to atmospheric oxygen. Spent nuclear fuel oxidation is a degradation or alteration mode that can significantly increase the potential radionuclide release rate in the potential repository. This results from the transformation of the UO_2 phase of spent nuclear fuel to a U_4O_9 lattice (slight volume decrease) and then to U_3O_8 (about a 30 percent volume expansion), increasing the surface area of the spent nuclear fuel exposed relative to the original pellet fragment or grain area. The U_3O_8 phase can split the zircaloy cladding lengthwise. Dry-bath weight gain tests are in progress to determine spent nuclear fuel oxidation response. These are long-term weight gain tests conducted in a hot cell. These tests primarily use low temperatures (less than 200°C) to examine oxidation rate, but one dry bath operates at 255°C to accelerate the oxidation rate. On the basis of information obtained from the dry-bath tests, thermogravimetric apparatus tests were initiated at a higher range of temperatures (250 to 320°C). These two types of tests will provide temperature-time-phase response as UO_2 spent nuclear fuel oxidizes to U_4O_{9+x}, then to U_3O_{8+x}, and finally to UO_3.

The dry-bath oxidation testing continued at a reduced level until January 1997, when a facility-wide electrical outage caused the tests to be shut down. The baths are being maintained in a manner where they can be restarted if additional ATM samples are obtained. Existing funding is being focused on thermogravimetric analysis testing.

Previous work to interpret the mechanisms of oxidation of spent nuclear fuel from the U_4O_9 ($UO_{2.4}$) state to the U_3O_8 phase has been complicated by extensive scatter in the data from thermogravimetric analysis. Detailed analysis has led to the probability that the scatter is caused by the difference in radial burnup in the samples coupled with the small (200 mg) sample size.

To test the hypothesis of a burnup effect on the oxidation of $UO_{2.4}$ to U_3O_8, nine of the previously oxidized samples of ATM-105 fuel were analyzed using thermal ionization mass spectrometry. The measured atom ratios were used in accordance with ASTM procedure E 321-79 "Standard Test Method for Atom Percent Fission in Uranium and Plutonium Fuel (Neodymium-148 Method)" to calculate the localized burnup of each individual sample. The estimated experimental uncertainty is ± 2.5 percent. All these samples, with the exception of one which came from the top of the fuel rod, came from the same 22 in. (56 cm) axial length segment of the same fuel rod. The axial gamma scans indicate a variation of only a few percent, but because of the small sample size used (~200 mg) coupled with the radial burnup distribution, a burnup distribution of 27.5 to 32.5 MWd/kgM within the samples tested was found.

More importantly, the results seem to verify the hypothesis of the burnup dependence of the oxidation kinetics from $UO_{2.4}$ to U_3O_8. In all instances (within uncertainty) at each temperature, the lower burnup fragments oxidized faster than the higher burnup samples. Also, it appears that for any measurable plateau of the oxygen-to-metal ratio to exist at 305°C, the burnup must be ~30 MWd/kgM. At 283°C, it appears that the length of time on the plateau increases with increasing burnup.

Two fragments from the radial center of ATM-104 Rod MKP-109 segment M underwent gamma energy analysis before being loaded in the thermogravimetric analysis. An attempt was made to minimize the amount of rim in each sample, although it is not possible with the equipment available to totally ensure that none of the outer rim is present in either sample.

Sample 12-3-96-104-01, a single fragment weighing 184.53 mg, was loaded in thermogravimetric analysis #1 and has operated for 650 hours (120 days on March 31) at 305°C. An oxygen-to-metal ratio of 2.40 was reached within 50 hours, and the sample remained on the plateau for approximately 450 hours. Thereafter, the sample has been slowly gaining weight and is currently at an oxygen-to-metal ration of ~2.44. This test continues to run. Sample 12-3-96-104-02, a single fragment weighing 213.90 mg, was loaded in thermogravimetric analysis #2 and also has operated for 650 hours (120 days on March 31) at 305°C. The first plateau at an oxygen-to-metal ratio of ~2.39 was reached within 50 hours; this sample is still on the plateau and the test continues to run.

The nominal pellet average burnup for the ATM-104 fuel in this region is 44 MWd/kgM, higher than the ~30 MWd/kgM for the ATM-105 fuel used in previous tests. The higher burnup ATM-104 fuel was hypothesized to have a longer plateau, or at least a longer time to form U_3O_8, than the ATM-105 fuel. The ATM-105 fragments oxidized previously at 305°C exhibited no extended plateau behavior; the one sample taking the longest to oxidize to an oxygen-to-metal ratio above 2.4 loosely displayed a plateau for about 50 hours. The current runs, with a minimum plateau of 450 hours, seem to verify the hypothesis that higher burnup fuel stabilizes the $UO_{2.4}$ matrix and delays the onset of U_3O_8 formation.

Forecast: Dry-bath testing will continue. Operation of the two thermogravimetric analysis systems and the analysis of the data will continue. Future tests, in accordance with the approved test plan addendum, include oxidation at 305°C of ATM-108 (gadolinium-doped) fuel, and fresh fragments of ATM-105 fuel cut to differentiate the radial surface and center fragments.

Subactivities 1.5.2.1.3 through 1.5.2.1.6. No progress was made during this reporting period on the four subactivities that address (1) the corrosion of zircaloy, (2) the corrosion of and radionuclide release from other materials, (3) the evaluation of the inventory and release of carbon-14 from the zircaloy cladding, and (4) other experiments on the spent nuclear fuel waste form.

6.10.3 Performance Assessment Activity 1.5.2.2 - Characterization of the Glass Waste Form

The objective of this activity is to provide the data required to calculate radionuclide release rates from glass waste forms.

Subactivity 1.5.2.2.1 - Leach testing of glass. Long-term unsaturated tests (drip tests) of two glass compositions (Savannah River defense waste processing facility and West Valley ATM-10) continued in two test series labeled the N2 and N3 series. These tests are being used to determine the types and quantities of radionuclide elements released from waste glasses when subjected to an intermittent dripping water contact scenario. Both soluble and colloidal radionuclide releases of actinides and technetium are being measured. A 304L stainless steel sample holder is also present in these tests to simulate the presence of the pour canister material on glass waste form behavior.

The defense waste processing facility glass is being tested in the N2 test series, which has been in progress for 568 weeks (10.9 years) as of March 31, 1997. The tests were sampled on January 23, 1997, as scheduled. Preliminary evaluation of solution analyses from these tests shows that the #10 test continues to release plutonium and americium at a rate substantially greater than the rates of tests #9 and #12, while the release of soluble elements is comparable in all the tests. This observation is consistent with colloidal release of the actinides in test #10 (Fortner et al., 1996).

The West Valley glass (ATM-10) is being tested in the N3 test series, which has been in progress for 493 weeks (9.5 years) as of March 31, 1997. The tests were successfully sampled on January 13, 1997, as scheduled. Inductively coupled plasma-mass spectroscopy data for solution analyses from these tests indicates the elemental release rates are maintaining steady values similar to those provided in Progress Report #15 (DOE, 1997e).

For both of the N2 and N3 samples, transuranic entrainment by colloids and particulates is being determined by ultracentrifugation-filtration of the test solutions, which removes materials with dimensions greater than about 0.1 micron. Inductively coupled plasma-mass spectroscopy analysis of cation concentrations in the solutions from both the N2 and N3 tests is in progress. No further analytical work on either of these sample sets will be performed.

These tests are providing data on radionuclide release mechanisms and degradation/alteration rates of glass waste forms. These data are being used to constrain and guide on-going model development for glass corrosion. Additional glass waste form testing will be required to model glass degradation/alteration modes and the subsequent dissolution/release of radionuclides as soluble and colloidal species over the range of potential repository environments.

Subactivity 1.5.2.2.2 - Materials interactions affecting glass leaching. The N2 and N3 test series contain stainless steel holders that simulate the presence of the 304L pour canister that will be present in the repository. Colloidal-sized iron particles have been identified in the N2 and N3 test solutions. This is important because sorption of radionuclides onto these particles provides a transport mechanism for radionuclides that is not solubility controlled. No other work in this subactivity is in progress.

Forecast: Degradation/alteration testing of the glass waste form in these long-term unsaturated tests will continue. Limited analysis of colloids from previous tests will continue.

Subactivity 1.5.2.2.3 - Cooperative testing with waste producers. No progress was made during the reporting period; this was an unfunded activity.

6.10.4 Performance Assessment Activity 1.5.3.1 - Integrate Scenarios for Release From Waste Package

The objective of this activity is to first identify the features, events, and processes and then to develop the scenarios made up of these features, events, and processes for which the engineered barrier system subsystem must be analyzed in accordance with 10 CFR 60.113. This paragraph of 10 CFR Part 60 sets quantitative performance requirements to be satisfied by the waste package and engineered barrier system, assuming anticipated processes and events. Thus, this activity must identify which features, events, and processes and scenarios are expected.

Subactivities 1.5.3.1.1 through 1.5.3.1.4. No progress was made during the reporting period; this was an unfunded activity. The current emphasis for the total system performance assessment for the viability assessment is on total system performance scenarios, rather than on subsystem performance. The four subactivities cover: (1) developing scenario identifications, (2) separating scenarios into anticipated and unanticipated categories, (3) developing parameters of the scenarios, and (4) comparing the anticipated scenarios with the design envelope of the waste package. The identification and evaluation of waste package design basis accidents is addressed in Section 5.1.3 of this progress report. Analyses reported in Sections 5.2.1 and 5.2.3 of this progress report will contribute to the technical basis for developing the scenarios for the next total system performance assessment. Submodel development in Section 6.10.8 of this progress report will use and possibly extend work on the scenarios reported in Section 6.10.4 in Progress Report #13 (DOE, 1996f).

Forecast: Improved analysis of engineered barrier system scenarios will become possible from developments in Sections 6.10.8 and 6.10.10 of this progress report, although the near-term emphasis of those activities will be on release-rate modeling to support total system performance assessment for the viability assessment. Engineered barrier system scenarios will be incorporated in the sensitivity studies and scenario development for the total system performance assessment for the viability assessment.

6.10.5 Performance Assessment Activity 1.5.3.2 - Develop Geochemical Speciation and Reaction Model

The objective of this activity is to further develop the geochemical modeling code EQ3/6 for use in modeling of waste form radionuclide release and the behavior of the released radionuclides.

Subactivity 1.5.3.2.1 - Develop data base for geochemical modeling. GEMBOCHS was augmented by including reference-state thermodynamic data and heat-capacity coefficients for a large number of cadmium, hafnium, lead, titanium, zinc, and zirconium species not previously available in GEMBOCHS. In aggregate, these data expand significantly the compositional range of problems that can be addressed using geochemical modeling software; they permit improved modeling capabilities for both direct and analog geochemical scenarios relevant to the potential repository at Yucca Mountain. Numerous relatively small-scale additions and revisions have also been incorporated into the data base and associated software. New thermodynamic data file suites that incorporate all the foregoing improvements have been generated for use with the geochemical software packages EQ3/6 and GWB (The Geochemist's Workbench) and are available to Project participants via anonymous Internet file transfer protocol (ftp) access.

In support of modeling studies associated with altered zone characterization (see Section 3.14 of this progress report), graphical software was augmented to facilitate generation of GEMBOCHS-derived thermodynamic data files for use with the reactive-transport software package OS3D/GIMRT (Heefel and Yabusaki, 1995). Several such files were generated and have been used extensively by those involved with altered zone modeling.

Existing thermodynamic and solubility data for important radionuclides (americium, nickel, neptunium, plutonium, technetium, uranium, and zirconium) at both ambient and elevated temperatures are being evaluated with particular attention paid to solid phases that may control solubilities under expected repository conditions.

Forecast: The GEMBOCHS data base and software library will continue to be improved by including new and revised thermodynamic data and extrapolation algorithms as they become available in the scientific literature. These improvements will be incorporated into revised EQ3/6, GWB, and OS3D/GIMRT data file suites, which will be available via anonymous Internet file transfer protocol (ftp) access.

A limited number of new data will be collected so that a recommendation can be made on the ability to extrapolate thermodynamic properties of the key radionuclides to elevated temperatures. If extrapolation is not possible, a recommendation will be made assessing the risk the Project is assuming if thermodynamic properties are not measured at elevated temperatures. A recommendation will also be made assessing the risk the Project is assuming if solubilities are not measured at elevated temperatures.

A critical review of literature values for the thermodynamic data on nickel and zirconium will be conducted. These two elements are not being covered by the Nuclear Energy Agency reviews. The goal is to use the Nuclear Energy Agency critical review procedures to provide the Project with a qualified data base for the two radionuclides with priority one data needs that are not being addressed by the Nuclear Energy Agency or Project data collection efforts.

Subactivity 1.5.3.2.2 - Develop geochemical modeling code. The objective of this task is to develop the geochemical modeling software EQ3/6 (Daveler and Wolery, 1992; Wolery, 1992a and b; Wolery and Daveler, 1992), which provides capabilities for analyzing and simulating interactions among water, rock, nuclear waste, and other repository components in the near-field environment, the altered zone, and the far-field environment. Qualified versions of the software are being maintained in the Version 7 line, and additional modeling capabilities were added in previous reporting periods to the Version 8 line. EQ3/6 is supported on both UNIX workstations and 386/486/Pentium personal computers.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.10.6 Performance Assessment Activity 1.5.3.3 - Generate Models for Release From Spent Fuel

The objective of this activity is to develop models for the oxidation, degradation, dissolution, and radionuclide release rate response of spent nuclear fuel waste forms.

Subactivity 1.5.3.3.1 - Generate release models for spent nuclear fuel. The primary transport mode of radionuclides from spent nuclear fuel waste forms is by water contacting, wetting, dissolving, and reacting at the surface of the spent nuclear fuel. The development of a model to describe the physical, chemical and radiolytic processes requires several submodeling steps. These include a submodel to predict the oxidation state of the fuel (UO_2, U_4O_{9+x}, U_3O_8 or UO_3). Given the oxidation state, the transport equation, which is the mass balance equation, is the basis for developing release submodels that depend on degradation and dissolution rates.

UO_2 spent nuclear fuels oxidize to higher uranium oxide phases in an oxygen atmosphere. The oxidation response of spent fuels affects the radionuclide release in potential repository environments because of two independent functional consequences of the higher oxides. The first effect is geometrical and results from the surface area and volume changes that occur as the higher oxides form. The second effect results from the higher dissolution rate of the U_3O_8 oxide and the UO_3 oxide hydrates.

The basis of spent fuel oxidation response model development depends strongly on experimental data obtained from thermogravimetric analysis and oven dry bath oxidation testing methods (see Sections 6.9.6 and 6.10.2 of this progress report). The models provide response functions for the elapsed time to higher oxidation phases as a function of temperature and nominal grain size and for the quantity (volume) of a higher oxidation phase as a function of time, temperature, and nominal grain size. The two spent nuclear fuel oxidation phase responses examined thus far are the UO_2 to U_4O_9 phase transformation and the U_4O_9 to U_3O_8 phase transformation. For these two-phase transformations, oxidation response models have been developed. These models were developed independently from dry oxidation data obtained at 255°C. A combination of these models was applied to predict in a bounding manner the 255°C dry-bath data. The predicted results of the model using several distinct grain sizes were compared with the 255°C data. The comparisons show that the experimental results are bounded by the models at small grain sizes. A refinement to this response model that uses a distribution of grain sizes would be more nominally representative of the actual test data. These models are simplistic in form, based on limited data, but useful for the current stage of design and performance assessment analyses. The oxidation models and their comparison with data at 255°C were included in the recent revision of Section 3.2.2 of the Waste Form Characteristics Report, version 1.2 (Stout and Leider, 1996).

The framework for developing dissolution response models is nonequilibrium thermodynamics (Stout, 1996). Three different function forms have been developed to describe the dissolution response of unoxidized UO_2 spent nuclear fuel waste forms. The first form is a classical Onsager relationship (deGroot and Mazur, 1962) for the dissolution rate, which is linearly coupled to the energy change of the solid dissolving into a liquid and the diffusional mass flux energy changes. This is expected to describe the dissolution response close to thermodynamic equilibrium. In addition to this near-equilibrium Onsager model, two nonequilibrium models were derived for dissolution reaction processes occurring far from thermodynamic equilibrium. In the first model, a functional form was derived that is similar to the classical Butler-Volmer relationship (Bockris and Khan, 1993) used in electrochemical corrosion processes. This model has an exponential representation for the energy changes, chemical plus electrochemical potentials, occurring during dissolution across a solid-liquid interface. A second model was derived in this reporting period and is referred to as the second Butler-Volmer model. Logarithmic dependent chemical potentials were substituted in this model, which is similar to the classical chemical kinetic rate law (Stumm and Morgan, 1981) for gaseous and liquid chemical reaction processes. These nonequilibrium models provide a chemical thermodynamic basis for spent fuel dissolution models.

Regression analysis of the set of unirradiated UO_2 and spent UO_2 nuclear fuel flow-through dissolution rate data provided a satisfactory fit to the two Butler-Volmer models. In addition to temperature and the water chemistry variables, burnup is included as a surrogate for fission product concentrations. For regression purposes, the Butler-Volmer models used a quadratic polynomial for chemical potential energy changes. In this reporting period, the second Butler-Volmer model was used for regression analyses, and used a polynomial of logarithmic terms for chemical potential energy changes. Some interaction and quadratic terms were included with both models to improve the fit. These features make both models nonlinear, because the chemical potential energy change terms are in the exponential function and contain quadratic terms. This second Butler-Volmer model provides better dissolution rate estimates compared with the first form, particularly when it is extrapolated to the less alkaline and reactive pH range of 7 to 8. Additional dissolution data for higher burnup fuels are needed to increase the reliability of the models. The function forms and the results of the regression analyses were incorporated in the recent revision of Section 3.4.2 of the Waste Form Characteristics Report, version 1.2 (Stout and Leider, 1996).

Model development to describe release rates on the basis of the unsaturated test data continued. The unsaturated test data indicated that significant alteration of the fuel surface has occurred during the approximately four years of testing, and the surfaces of grains are probably exposed and wetted by a thin water film. The model development assumes quasi-steady rate processes are occurring for dissolution rates, precipitation rates, colloidal kinetics, and adsorption kinetics. Using this assumption, the mass balance equation for a generic species can be simplified and reduced to a mass transport expression that strongly depends on water flowing over a wetted area. For unsaturated flows, this results in a weak dependence on the total surface of spent fuel exposed in that only the wetted surface contributes to the release rate. Using the high and low drip-rate test data, preliminary values for a length scale parameter are being determined.

Forecast: Development of submodels for release rate terms in the mass transport model will continue. Submodels will be refined that are consistent with the limited subsets of dissolution/release data. To include known effects of UO_2 spent nuclear fuel oxidation on release rates, updates, refinements, and impacts of the current oxidation models will be completed as additional thermogravimetric and oxidation dry-bath data become available.

6.10.7 Performance Assessment Activity 1.5.3.4 - Generate Models for Release From Glass Waste Forms

Subactivity 1.5.3.4.1 - Generate release models for glass waste forms. Chemical modeling of glass degradation is being used to synthesize results from on-going 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.

Current work is focused on examining the effect of dissolved cations such as aluminum, iron, magnesium and others on glass waste form performance, with most of the work concentrating on magnesium. Previous experimental data suggest that dissolved magnesium can have a significant beneficial effect on glass durability. To better understand the mechanism by which dissolved magnesium enhances glass durability, and to quantify the effect, glass dissolution rates are being measured in flow-through tests where the pH buffer fluids have been doped with magnesium. The tests will begin in the next reporting period.

A literature review of available data on the effects of dissolved species on glass durability was completed and written up. This review has been incorporated into the current version of the Waste Form Characteristics Report, version 1.2 (Stout and Leider, 1996).

Forecast: The theoretical and experimental analysis of the effect of dissolved species on glass waste form durability will continue. The flow-through tests with dissolved magnesium present will be completed, and another set examining the effect of dissolved iron started. Results of these experiments will provide parameters that can be incorporated into the glass dissolution model and used to model other available experimental data for model validation.

6.10.8 Performance Assessment Activity 1.5.3.5 - Waste Package Performance Assessment Model Development

The objectives of this activity are to (a) integrate submodels of processes that affect radionuclide release from waste packages into a system model for waste package performance (Subactivity 1.5.3.5.1), (b) develop a method for providing probability distributions for individual waste package and ensemble performance that incorporate uncertainties in conditions and waste package parameters (Subactivity 1.5.3.5.2), and (c) determine experimentally what fraction of the water dripping onto a container would enter the container through a breach in the container wall (Subactivity 1.5.3.5.3).

Performance Assessment Model Abstractions

Work is reported in Sections 6.8.4, 6.8.6, and 6.8.7 of this progress report.

Glass Alteration Submodel

A glass alteration model has been developed for use in the subsystem performance assessment model, which is reported in the Waste Form Characteristics Report, Version 1.2 (Stout and Leider, 1996). The model covers water contact modes of trickle flow over the glass surface and gradual immersion in the container in a bathtub mode by a slow inflow of water. The model is based on results of a matrix of batch tests with sudden immersion in a fixed quantity of water and on flow through tests at relatively high water flow rates. Attention to the experimental tests shows dependence on temperature, pH, and dissolved silica content in the water. The silica content increases with the glass dissolution, hence the model tracks the changing silica content during the water contact in each mode. Another corollary is that the fraction of silica reprecipitated is important in determining the amount of silica remaining in solution.

Subactivity 1.5.3.5.2 - Development of uncertainty methodology. No progress was made during the reporting period; this was an unfunded activity.

Subactivity 1.5.3.5.3 Water flow into and out of a breached container. No progress was made during the reporting period; this was an unfunded activity.

Forecast: Enhanced subsystem models will be developed during the next reporting period for intended use in model-testing sensitivity analyses and in the total system performance assessment for the viability assessment. The development will be guided by plans being developed in and subsequent to the workshops held this reporting period.

6.10.9 Performance Assessment Activity 1.5.4.1 - Deterministic Calculation of Releases from the Waste Package

The objective of this activity is to use the waste package system model developed in Performance Activity 1.5.3.5 (Section 6.10.8 of this progress report) to predict the performance of waste packages in given environmental time histories for analysis of sensitivity to environmental parameters and design alternatives, for use within the probabilistic calculation of the perform. ce of the waste packages in the potential repository, and for use in total system performance assessment analyses.

No calculations in this category were carried out. Instead, effort was on a cycle of improvements to the models; see Section 6.10.8 of this progress report.

Forecast: See Section 6.10.10 of this progress report.

6.10.10 Performance Assessment Activity 1.5.4.2 - Probabilistic Calculation of Releases from the Waste Package

The objective of this activity is to provide a probabilistic analysis of waste package performance addressing uncertainties in the waste package environment and components and to provide the probability distribution of radionuclide release rates for use in SCP Issue 1.1 (Section 6.13 of this progress report), using the uncertainty model developed in Performance Assessment Activity 1.5.3.5 (Section 6.10.8 of this progress report).

No calculations in this category were carried out. Instead, effort was on a cycle of improvements to the models; see Section 6.10.8 of this progress report.

Forecast: Scoping calculations using prototype advancements of submodels will be performed as model testing and sensitivity analyses in support of the model development; see Section 6.10.8 of this progress report. Results will be used as feedback to the subsystem model development and scenario development for the total system performance assessment for the viability assessment

6.10.11 Performance Assessment Activity 1.5.5.1 - Determine Radionuclide Transport Parameters

The objective of this activity is to measure the distribution, transport, and interaction of actinides and fission products in the engineered barrier system and near-field environment materials. Engineered barrier system and near-field environment materials will be subjected to contact with radionuclide-bearing solutions under a variety of conditions to identify the mode of materials interaction and radionuclide transport and to derive parameters that can be used to bo d the radionuclide source term.

Estimating the radionuclide source term requires a combination of experimental and modeling tasks designed to assess both the release of radionuclides from the waste form and their transport through and interaction with the components of the engineered barrier system and near-field environment. Integrated testing is planned to bound the flux of radionuclides that pass through the engineered barrier system and near-field environment. This activity will measure and model (a) the potential for transport of radionuclide elements from the waste form through the introduced materials, (b) their alteration products, and (c) the altered host rock that will make up the post-emplacement near-field environment of the potential repository at Yucca Mountain. Experimental measurements will be combined with conceptual and mechanistic models to bound the concentrations of radionuclide elements that will be released from the engineered barrier system and near-field environment. The estimated releases will be used in the performance assessment of the engineered barrier system and near-field environment subsystem to calculate the radionuclide source term to be used in total system performance assessment.

Subactivity 1.5.5.1.1 - Radionuclide distribution in tuff wafers. No progress was made during the reporting period; this activity was incorporated in Subactivity 1.5.5.1.3 (see below).

Subactivity 1.5.5.1.2 - Radionuclide distribution in tuff cores. No progress was made during the reporting period; this activity was incorporated in Subactivity 1.5.5.1.3 (see below).

Subactivity 1.5.5.1.3 - Determine radionuclide transport parameters necessary to define the source term. Experiments began to determine the transport of uranium and neptunium through waste package corrosion products and hydrothermally altered concrete.

In one set of experiments, water-jacketed chromatography columns were set up containing mixtures of fine-grained quartz and hematite to simulate transport of radionuclides through corrosion products. Conservative (iodine) and adsorbing (uranium and neptunium) tracers are being passed through these columns at pHs between 4 and 8 and at temperatures between ambient and 80°C.

In the other set of experiments, crushed material and fractured cores from the ESF concrete invert were placed in hydrothermal vessels and treated at 200°C to prepare them for radionuclide transport experiments. For these experiments, the crushed material will be placed in chromatography columns and the fr[illegible]red cores will be placed in a core-flow device.

Forecast: The experimen[illegible]asurement of iodine, uranium, and neptunium transport through a hematite/quartz mixture and through hydrothermally altered crushed and fractured concrete will continue. The results will be used to define model parameters for predicting the radionuclide source term from the engineered barrier system.

6.10.12 Performance Assessment Activity 1.5.5.2 - Radionuclide Transport Modeling in the Near-Field Waste Package Environment

The objective of this activity is to use the flow and transport model for hydrologic representation of the near-field host rock developed in Design Studies 1.10.4.1 through 1.10.4.5 (Sections 5.2.2 through 5.2.6 of this progress report). The model will be validated using data from integrated testing activities and tracer tests planned in the ESF.

Subactivity 1.5.5.2.1 - Validation of near-field transport model using laboratory and field experimental data. Near-field transport of waste package releases was modeled using the total system performance assessment computer code RIP (Golder, 1995) in support of the Engineered Barrier System Performance Requirements Study (CRWMS M&O, 1996bb), which is discussed in Section 4.1.19 of this progress report.

Forecast: No work is currently planned for FY 1997. Development of engineered barrier system transport scenarios and model requirements may evolve, however, from the performance assessment abstraction-testing workshop on waste form alteration and radionuclide mobilization (see Analysis Plan 6 in Section 6.8.6 of this progress report).

Subactivity 1.5.5.2.2 - Application of near-field transport model to waste package releases. Mechanistic coupled flow and transport models as implemented in the computer modeling codes OS3D/GIMRT (Steefel and Yabusaki, 1995) and X1t have been used to define transport experiment protocols and to make predictions of transport experiment results.

Forecast: Experimental results will be compared with model predictions Mechanistic models will be applied to bound transport of selected radionuclides through engineered barrier system and near-field environment materials. A simplified model abstraction of transport through near-field environment and engineered barrier system materials will be developed for use in the total system performance assessment for the viability assessment.

6.11 SEAL PERFORMANCE (SCP SECTION 8.3.5.11)

SCP Section 8.3.5.11 addresses whether the design of the seal system will meet the requirements of 10 CFR 60.134(a) and (b) and how seal performance will contribute to the engineered barrier system performance in accordance with 10 CFR 60.113(a)(1). The seal system is defined as being composed of shafts, ramps, exploratory boreholes and their seals, and the sealing components of the underground facility.

No progress was made during this reporting period, this was an unfunded activity. See Section 4.5 of this progress report for related information.

Forecast: No work is planned for FY 1997.

6.12 GROUND-WATER TRAVEL TIME (SCP SECTION 8.3.5.12)

SCP Section 8.3.5.12 addresses Issue 1.6, which asks whether the site will meet the performance objective for pre-waste-emplacement ground-water travel time as required by 10 CFR 60.113.

A series of workshops were held to address the abstraction of process-level models for total system performance assessments. During this reporting period, ground-water travel time and related topics were addressed in three of these workshops:

1. Unsaturated zone flow workshop (Section 6.8.3)
2. Thermohydrology workshop (Section 6.8.7)
3. Unsaturated zone radionuclide transport workshop (Section 6.8.5).

Two more workshops addressing ground-water travel time and related topics will be held in the next reporting period:

1. Saturated zone flow and radionuclide transport workshop
2. Biosphere workshop.

Descriptions of the workshops held to date are included in Section 6.8 of this report.

6.12.1 Performance Assessment Activity 1.6.2.1 - Model Development

The objective of this activity is to develop calculational models for predicting pre-waste-emplacement ground-water travel time.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.12.2 Performance Assessment Activity 1.6.2.2 - Verification and Validation

The objective of this activity is verification of computer codes and validation of mathematical models for ground-water travel time analyses.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997

6.12.3 Performance Assessment Activity 1.6.3.1 - Analysis of Unsaturated Flow System

The objective of this activity is to determine which flow paths or sets of flow paths of likely radionuclide travel in the unsaturated zone will be used in ground-water travel time calculations.

Work during this reporting period concentrated on the preparation, supporting analyses, and evaluations for the unsaturated zone flow, unsaturated zone transport, and thermohydrology workshops (see Section 6.8 of this report for details).

Forecast: Modeling of heterogeneous inflow into drifts will continue to provide input to performance assessment of flow and transport processes on the drift scale. The episodic pulse infiltration will be systematically quantified. The stochastic continuum model and other heterogeneous models will be used in additional sensitivity analyses to account for the spatial and temporal variability of percolation into the repository drifts.

6.12.4 Performance Assessment Activity 1.6.4.1 - Calculation of Pre-Waste-Emplacement Ground-Water Travel Time

The objective of this activity is to define performance measures and to perform related analyses of pre-waste-emplacement ground-water travel time.

No progress was made on any of the three subactivities of this activity during the reporting period: Subactivity 1.6.4.1.1—Performance allocation for Issue 1.6, Subactivity 1.6.4.1.2—Sensitivity and uncertainty analyses of ground-water travel time, and Subactivity 1.6.4.1.3—Determination of the pre-waste-emplacement ground-water travel time. This was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.12.5 Performance Assessment Activity 1.6.5.1 - Ground-Water Travel Time after Repository Construction and Waste Emplacement

The objective of this activity is to predict the ground-water travel time to the water table using the hydrologic properties changed as a result of repository construction and waste emplacement. This is needed for comparing pre- and post-emplacement ground-water travel times to establish the extent of the disturbed zone. This objective includes developing the model domain, conceptual flow models, computational codes, model parameters, and boundary conditions to model ground-water flow at Yucca Mountain influenced by thermally driven alterations to the hydrology.

Work concentrated on the preparation, supporting analyses, and evaluations for the unsaturated zone flow, unsaturated zone transport, and thermohydrology workshops (see Section 6.8 of this report for details).

Forecast: Work will continue using the three-dimensional site-scale model grid to evaluate the effects of thermal loading on ground-water travel times and the effects of three-dimensional geologic features, such as faults, on the ambient moisture, gas and heat flow in the Yucca Mountain.

6.12.6 Performance Assessment Activity 1.6.5.2 - Definition of the Disturbed Zone

The objective of this activity is to re-evaluate the definition of the disturbed zone.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.13 TOTAL SYSTEM PERFORMANCE (SCP SECTION 8.3.5.13)

SCP Section 8.3.5.13 addresses Issue 1.1, which asks whether the MGDS will meet the system performance objective for limiting radionuclide releases to the accessible environment as required by 10 CFR 60.112 and 40 CFR 191.13.

See Section 6.8 of this progress report for related work.

6.13.1 Performance Assessment Activity 1.1.2.1 - Preliminary Identification of Potentially Significant Release Scenario Classes

The objective of this activity is to preliminarily identify significant release scenario classes for the purpose of determining data and information needs that must be supplied by the Yucca Mountain site characterization program.

No progress was made on any of the two subactivities of this activity during the reporting period: Subactivity 1.1.2.1.1—Preliminary identification of potentially significant sequences of events and processes at the Yucca Mountain repository site and Subactivity 1.1.2.1.2—Preliminary identification of potentially significant release scenario classes. This was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.13.2 Performance Assessment Activity 1.1.2.2 - Final Selection of Significant Release Scenario Classes to be Used in Licensing Assessments

The objective of this activity is to use data and information obtained in the Yucca Mountain site characterization program to modify, if necessary, the set of significant release scenario classes developed in Performance Assessment Activity 1.1.2.1 (Section 6.13.1 of this progress report) and provide information for Performance Assessment Activities 1.1.3.1, 1.1.4.1, and 1.1.4.2 (Sections 6.13.3 through 6.13.5 of this progress report) in the preliminary phases of work.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.13.3 Performance Assessment Activity 1.1.3.1 - Development of Mathematical Models of the Scenario Classes

The objective of this activity is to construct mathematical models of the scenario classes developed in Performance Assessment Activities 1.1.2.1 and 1.1.2.2 (Sections 6.13.1 and 6.13.2 of this progress report).

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.13.4 Performance Assessment Activity 1.1.4.1 - The Screening of Potentially Significant Scenario Classes Against the Criterion of Relative Consequences

The objective of this activity is to identify the set of scenario classes representing the significant events and processes mentioned in 10 CFR 60.112 and 60.115.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.13.5 Performance Assessment Activity 1.1.4.2 - The Provision of Simplified, Computationally Efficient Models of the Final Scenario Classes Representing the Significant Processes and Events Mentioned in Proposed 10 CFR 60.112 and 60.115

The objective of this activity is to provide the simplified, computationally efficient models of the final scenario classes representing the significant events and processes mentioned in 10 CFR 60.112 and 60.113.

No progress was made during the reporting period, this was an unfunded activity.

Forecast: No work is planned for FY 1997

6.13.6 Performance Assessment Activity 1.1.5.1 - Calculation of an Empirical Complementary Cumulative Distribution Function

The objective of this activity is to construct an efficient, total system simulator that is capable of providing probabilistic estimates of radionuclide releases to the accessible environment, under both nominal and disturbed conditions, for 10,000 years after repository closure.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.14 INDIVIDUAL PROTECTION (SCP SECTION 8.3.5.14)

SCP Section 8.3.5.14 addresses Issue 1.2, which asks whether the MGDS will meet the requirements for limiting individual radiation doses in the accessible environment for 1000 years after waste disposal as required by 40 CFR 191.15.

See Section 6.8 of this progress report for related work.

6.14.1 Performance Assessment Activity 1.2.1.1 - Calculation of Doses Through the Ground-Water Pathway

The objective of this activity is to use the methodology developed for total system performance assessment (Section 6.13 of this progress report) to calculate the radionuclide transport to the boundary of the controlled area and radiation doses from drinking contaminated ground water during the first 1000 years after waste disposal.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.14.2 Performance Assessment Activity 1.2.2.1 - Calculation of Transport of Gaseous Carbon-14 Dioxide Through the Overburden

The objective of this activity is to estimate the transport time for gaseous carbon-14 dioxide from the potential repository to the land surface during the first 1000 years after waste disposal.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.14.3 Performance Assessment Activity 1.2.2.2 - Calculation of Land-Surface Dose and Dose to the Public in the Accessible Environment Through the Gaseous Pathway of Carbon-14

The objectives of this activity are to collect the necessary data on carbon-14 inventory and meteorology and to calculate upper-bound values for external and internal radiation doses. Internal radiation doses include doses from both inhalation and ingestion.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.15 GROUND-WATER PROTECTION (SCP SECTION 8.3.5.15)

SCP Section 8.3.5.15 addresses Issue 1.3, which asks whether the MGDS will meet the requirements for the protection of special sources of ground water for 1000 years after waste disposal as required by 40 CFR 191.16.

See Section 6.8 of this progress report for related work.

6.15.1 Performance Analysis 1.3.1.1 - Determine whether any Aquifers near the Site Meet the Class I or Special Source Criteria

The objective of this activity is to determine whether any aquifers within the controlled area or within 5 km of the controlled area meet the Class I criteria as defined by the U.S. Environmental Protection Agency Ground Water Protection Strategy of 1984 (EPA, 1984) or special source criteria as defined by 40 CFR 191.12.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.15.2 Performance Analysis 1.3.2.1 - Determine the Concentrations of Waste Products in any Special Source of Ground Water during the First 1000 Years After Disposal

The objective of this analysis is to calculate the concentration of waste products in any special-source aquifers during the first 1000 years after waste disposal.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.16 PERFORMANCE CONFIRMATION (SCP SECTION 8.3.5.16)

This section of the progress report addresses SCP Section 8.3.5.16, Issue 1.7, on performance confirmation, which is directly related to the performance confirmation program requirements of 10 CFR 60.137.

The Performance Confirmation Plan is in preparation (CRWMS M&O, in prep.[h]). The plan will be based on the Performance Confirmation Concepts Study Report (CRWMS M&O, 1996z) completed in the previous reporting period and revised this period. The Performance Confirmation Plan will define the activities necessary to conduct the Performance Confirmation Program as specified in 10 CFR Part 60 Subpart F. This plan will specify those monitoring, testing, and analysis activities to be conducted for evaluating the accuracy and adequacy of the information to be used in the license application to determine that the waste isolation performance objectives for the period after permanent repository closure will be met.

The Performance Confirmation Program objectives are to: (a) confirm that subsurface conditions encountered and changes in those conditions during construction and waste emplacement during construction and waste emplacement operations are within the limits assumed in the license application; (b) confirm that natural and engineered systems and components that are required for repository operations, or that are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated; (c) evaluate compliance with regulatory and license requirements, related to postclosure performance requirements; and (d) evaluate the repository readiness for permanent closure.

The performance confirmation approach includes six major steps. The first step will define a performance confirmation baseline. This baseline will identify what processes and parameters are important to postclosure performance. The second step will predict values and variations of critical performance measures for the parameters in the performance confirmation baseline. These predictions will establish what is expected to be seen during construction and operations. The third step will establish tolerances or acceptable limits of deviations from predicted performance. The fourth step will monitor performance, perform tests, and collect data. The data will be analyzed and evaluated. Other evaluations will include process model validation, statistical tests, and total system performance assessment. Finally, if there are deviations from what was predicted or assumed, the Performance Confirmation Program will recommend and implement corrective actions. If the results indicate that postclosure regulatory performance requirements can be met with reasonable assurance, the Project will evaluate the repository readiness for permanent closure.

__Forecast__: The Performance Confirmation Plan will be completed at the end of FY 1997. The plan will supplement performance confirmation program requirements for the MGDS design. The plan will describe the performance confirmation program to be included in the License Application Plan. The Performance Confirmation Plan will contain a concept of operations, which will outline the operations necessary to conduct the performance confirmation program. The development of the performance confirmation program plan and the performance confirmation requirements will also facilitate the process of achieving compliance with the 10 CFR Part 60 requirement to begin performance confirmation during site characterization.

6.17 U.S. NUCLEAR REGULATORY COMMISSION SITING CRITERIA (SCP SECTION 8.3.5.17)

SCP Section 8.3.5.17 addresses Issue 1.8, which asks whether demonstrations for favorable and adverse conditions can be made as required by 10 CFR 60.122.

No progress was made during the reporting period; this was an unfunded activity.

Forecast: No work is planned for FY 1997.

6.18 HIGHER-LEVEL FINDINGS - POSTCLOSURE SYSTEM AND TECHNICAL GUIDELINES (SCP SECTION 8.3.5.18)

SCP Section 8.3.5.18 addresses Issue 1.9, that asks whether the high-level findings required by 10 CFR Part 960 can be made for the qualifying condition of the postclosure system guideline and the qualifying and disqualifying conditions of the technical guidelines for geonydrology, geochemistry, rock characteristics, climatic changes, erosion, dissolution, tectonics, and human interference.

No progress was made during the reporting period; this was an out-year activity. See Section 2.2.1 of this progress report for relevant regulatory activities.

Forecast: No performance assessment work is planned for FY 1997.

6.19 COMPLETED ANALYTICAL TECHNIQUES (SCP SECTION 8.3.5.19)

The purpose of this section is to identify analytical techniques and related process models and computer codes that were completed or acquired during this and prior reporting periods and that are being used or could be used to conduct pre- and postclosure performance assessment analyses.

The compilation in Table I-1, in Appendix I, is expected to provide a quick overview of available models and codes for performance assessments, site characterization, MGDS design, and related technical analyses. Progress with respect to model and code development, improvement, testing, verification, and validation during this reporting period is described in other sections of this progress report. Table I-2 identifies sections of this progress report where this work is described. Table I-1 includes mathematical models and computer codes that may need to be modified as new site information becomes available, as the engineered system design develops, as new understanding of natural and engineered barrier characteristics and processes is gained, and to demonstrate compliance with revised regulatory requirements. The listed codes and models may need additional verification and validation before they will be approved for use in a license application for an MGDS at the Yucca Mountain site.

Some of the listed computer codes are being used in support of MGDS design and Yucca Mountain site characterization and are not yet used in performance assessment analyses. They are listed because they could be used to support future performance assessment analyses. Some of the computer codes are not being used and funded at present. Not all computer codes are expected to be used in support of the license application. Also, a single table is presented rather than separate pre- and postclosure tables because some of the computer codes are used or could be used for both pre- and postclosure performance assessment analyses.

6.19.1 Postclosure Performance Assessment Analytical Techniques

See the preceding introductory text of Section 6.19 and Table I-1 in Appendix I of this progress report.

6.19.2 Preclosure Performance Assessment Analytical Techniques

See the preceding introductory text of Section 6.19 and Table I-1 in Appendix I of this progress report.

To support the design basis event pilot analyses, a simple Lotus 1-2-3 spreadsheet program was developed. This spreadsheet was used for consequence analysis for each design basis event evaluated to date. Use of the spreadsheet is expected to continue, with additional and/or improved process models being incorporated as available.

6.20 ANALYTICAL TECHNIQUES REQUIRING DEVELOPMENT (SCP SECTION 8.3.5.20)

The purpose of this section is to discuss (a) the need to develop analytical techniques for those areas where well-developed methods (as listed in Section 6.19 of this progress report) are currently not available and (b) the verification of computer codes and validation of mathematical models on which the methods are based.

SCP Section 8.3.5.20 lists the following five subsections: (1) analytical techniques, (2) plans for verification and validation, (3) verification of analytical techniques, (4) model validation, and (5) validation program. Descriptions of model and code development, improvement, testing, verification, validation, and applications during this reporting period, however, are covered both in other sections of this chapter and in other chapters. Table I-1 in Appendix I lists technical computer codes available in the Project; computer codes needing additional development and documentation are listed as "working versions." Table I-2, in Appendix I, lists sections of Progress Report #16 where activities are described that mention the computer codes listed in Table I-1. To establish a one-to-one correspondence between these two tables, all computer codes listed in Table I-1 are also listed in Table I-2, although some of them may not have been identified in other sections of this progress report.

6.21 SITE IMPACT EVALUATIONS

The objective of site impact evaluations is to identify any potentially adverse effects of site characterization activities on the postclosure waste isolation performance of a potential high-level radioactive waste repository at the Yucca Mountain site. These evaluations are documented in the determination of importance evaluations (see Section 2.3.2 of this progress report), which include test interference evaluations and which establish control requirements intended to limit adverse impacts of site characterization activities on waste isolation. Site impact evaluations are required by 10 CFR 60.15(c)(1), which states that "investigations to obtain the required information shall be conducted in such a manner as to limit adverse effects on the long-term performance of the geologic repository to the extent practical."

Site impact evaluations are performed to estimate the potential effects of site characterization activities, including the construction and operation of the ESF, on testing and the waste isolation capability of the Yucca Mountain site. Controls on activities are established to limit adverse effects, if needed, on the basis of these technical analyses. Evaluations during this reporting period included an analysis of subsurface water and materials use in the Thermal Testing Facility in the ESF. The previously established limits for underground water use (CRWMS M&O, 1997v) were re-interpreted for a new water use activity involving the drilling of a dense pattern of boreholes for instrumentation around the Thermal Testing Facility. Work concerning the use in the subsurface ESF and potential repository of committed concrete (i.e., concrete that will remain underground in a potential repository after closure) is being pursued in an attempt to bound the geochemical effects of such materials on postclosure performance.

6.21.1 Tracers, Fluids, and Materials (excluding water)

A strategy is being pursued to address potential postclosure performance issues concerning use of large quantities of committed cementitious materials within the potential waste-emplacement drifts (e.g., precast concrete lining segments). This strategy includes performance analyses of the consequences caused by geochemical effects of these materials reacting over geologic time and incorporating constraints on (a) the potential geochemical effects inside the drift (e.g., pH, mineral dissolution/precipitation); (b) the potential geochemical effects in the geosphere (e.g., cement fluid-driven rock alteration); (c) the impacts to performance parameters from the identified potential geochemical effects inside the drift (e.g., increased solubility limits, generation of colloids); and (d) the impacts to performance parameters from the identified potential geochemical effects in the geosphere (e.g., reduced radionuclide sorption, porosity/permeability changes). This work would consist mainly of summary reports of existing data and modeling studies of the processes involved. Work was begun in a number of these areas at the end of FY 1996.

At the request of the repository design group, performance assessment staff planned an overview study to address potential postclosure performance issues concerning use of large quantities of committed cementitious materials within the potential waste-emplacement drifts (e.g., precast concrete lining segments). Only the FY-1996 portion of this work was executed directly and was summarized in a status report (CRWMS M&O, 1996aa). This work focused on sensitivity studies to analyze the consequences of pH perturbations and on evaluating the importance of cement composition to the generation of geochemical effects. The second portion was conducted primarily by (a) a meeting of performance assessment, MGDS design, and site characterization personnel with outside experts on cements and (b) evaluating the reports resulting from that meeting. The status report provides details on both of these; below is a brief summary of the results of these activities.

Preliminary sensitivity analyses were conducted using current total system performance assessment models. These sensitivity analyses included the evaluation of enhanced solubility of radionuclides (americium, neptunium and plutonium) and of reduced sorption in the unsaturated zone (cases for no retardation for 10 m, 100 m, and the entire unsaturated zone) caused by alkaline fluids that may result from the use of cementitious materials. These analyses indicate that for migration of alkaline pH fluids, order-of-magnitude increases in peak radiation dose and substantial shifts to earlier times may result from the following:

- Negation of sorption throughout the entire unsaturated zone
- Increased concentrations of neptunium and plutonium
- Combination of no retardation for 10 or 100 m into the geosphere and increased concentrations of neptunium and plutonium (this case would most represent an alkaline plume migrating through the rock matrix).

On the basis of the information gathered, recommendations are preliminary at this time. However, if concrete is desirable as lining for ground support, then minimizing the potential postclosure impacts to the waste isolation capabilities of the site will most likely be achieved by (a) using precast concrete, (b) designing a mix with lower calcium to silicon ratio, (c) using techniques such as particle size engineering, steam-curing, or pressure-curing to reduce the concrete permeability and water content needed for higher silica cements, (d) using tuff aggregate, and (e) investigating alternative cements, such as C_2S (a calcium oxide/silicon dioxide cement), that may have a lower pH than standard Portland cements.

In addition to the above study, performance assessment personnel continued to integrate with the repository design group in order to progress in our ability to constrain geochemical effects from concrete including pH perturbations and from organic materials within the cements.

Forecast: Performance assessment support of repository design will continue as part of the performance assessment efforts in the determination of importance evaluations and near-field environment work. The cement system is being addressed by constructing a bounding model for the evolution of the material through time given potential repository conditions. Constraints on the conditions are being generated within the performance assessment near-field environment effort and simple models should be available by the end of FY 1997.

6.21.2 Water Management Criteria

Control of water use during site characterization is necessary to ensure that these activities do not affect some characteristic that may be important to the postclosure waste isolation performance of a potential repository. The control requirements and supporting analyses were documented in the determination of importance evaluations (see Section 2.3.2 of this progress report).

The primary focus of site impact evaluations for water management was the interpretation of existing subsurface water use analyses (CRWMS M&O, 1997v) as applied to activities in the Thermal Testing Facility (CRWMS M&O, 1996b). Analyses addressing the adverse effects on potential repository performance of water consumption (i.e., the water discharged and not recovered) were used to control the quantity and distribution of water consumption such that the identified adverse effects are negligible. The application of existing controls for underground water use, primarily for the tunnel boring machine and drill and blast excavation operations, were interpreted for wet drilling a dense pattern of instrumentation holes in the Thermal Testing Facility and for the effects of mobilization of in situ water from heater testing.

The revised analyses for subsurface water use (discussed in detail in Progress Report #15) have been issued (CRWMS M&O, 1997v).

Forecast: The evaluations for the ESF will have to be reworked or modified to address the effects of added water during construction of potential repository emplacement drifts. Continuing work concerning the effects on potential repository performance of committed concrete in the subsurface environment will be a part of the integration and abstraction-testing effort for both near-field geochemical and waste-form mobilization modeling. Work proposed for an east-west ESF drift may require new analyses depending on the location and potential uses of this excavation.

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