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

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[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)

<table>
<tr><th colspan="2">Formal Geologic Stratigraphy (after Sawyer et al., 1994)</th><th>Hydrogeologic Units (Modified from Montazer and Wilson, 1984)</th><th>Thermal/Mechanical Units (Ortiz et al., 1985)</th></tr>
<tr><td colspan="2">Qac</td><td>Alluvium</td><td>UO</td></tr>
<tr><td rowspan="10">Paintbrush Group</td><td>Tiva Canyon Tuff</td><td>Tiva Canyon Welded Unit TCw</td><td>TCw</td></tr>
<tr><td>pre-Tiva Canyon bedded tuff</td><td rowspan="5">Paintbrush Nonwelded Unit PTn</td><td rowspan="5">PTn</td></tr>
<tr><td>Yucca Mountain Tuff</td></tr>
<tr><td>pre-Yucca Mountain bedded tuff</td></tr>
<tr><td>Pah Canyon Tuff</td></tr>
<tr><td>pre-Pah Canyon bedded tuff</td></tr>
<tr><td rowspan="4">Topopah Spring Tuff</td><td rowspan="3">Topopah Spring Welded Unit TSw</td><td>TSw1</td></tr>
<tr><td>TSw2</td></tr>
<tr><td>TSw3</td></tr>
<tr><td rowspan="4">Calico Hills Nonwelded Unit CHn</td><td>CHn1v</td></tr>
<tr><td colspan="2">pre-Topopah Spring bedded tuff</td><td rowspan="2">CHn1z</td></tr>
<tr><td colspan="2" rowspan="2">Calico Hills Formation</td></tr>
<tr><td>CHn2z</td></tr>
<tr><td rowspan="6">Crater Flat Group</td><td rowspan="2">Prow Pass Tuff</td><td rowspan="6">Crater Flat Unit CFu</td><td>CHn3z</td></tr>
<tr><td>PPw</td></tr>
<tr><td rowspan="2">Bullfrog Tuff</td><td>CFun</td></tr>
<tr><td>BFw</td></tr>
<tr><td rowspan="2">Tram Tuff</td><td>CFMn</td></tr>
<tr><td>TRw</td></tr>
</table>

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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 this 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. Compare drift arrangements to ensure that they fit the geology and structure

4. Identify site geologic data requirements for determining usable area.

This design activity 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 feasibility 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 benefits 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/emplacement 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.)

Activity 1.12.2.1.2 - Hydraulic conductivity and consolidation testing of crushed tuff. 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 contribute 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.

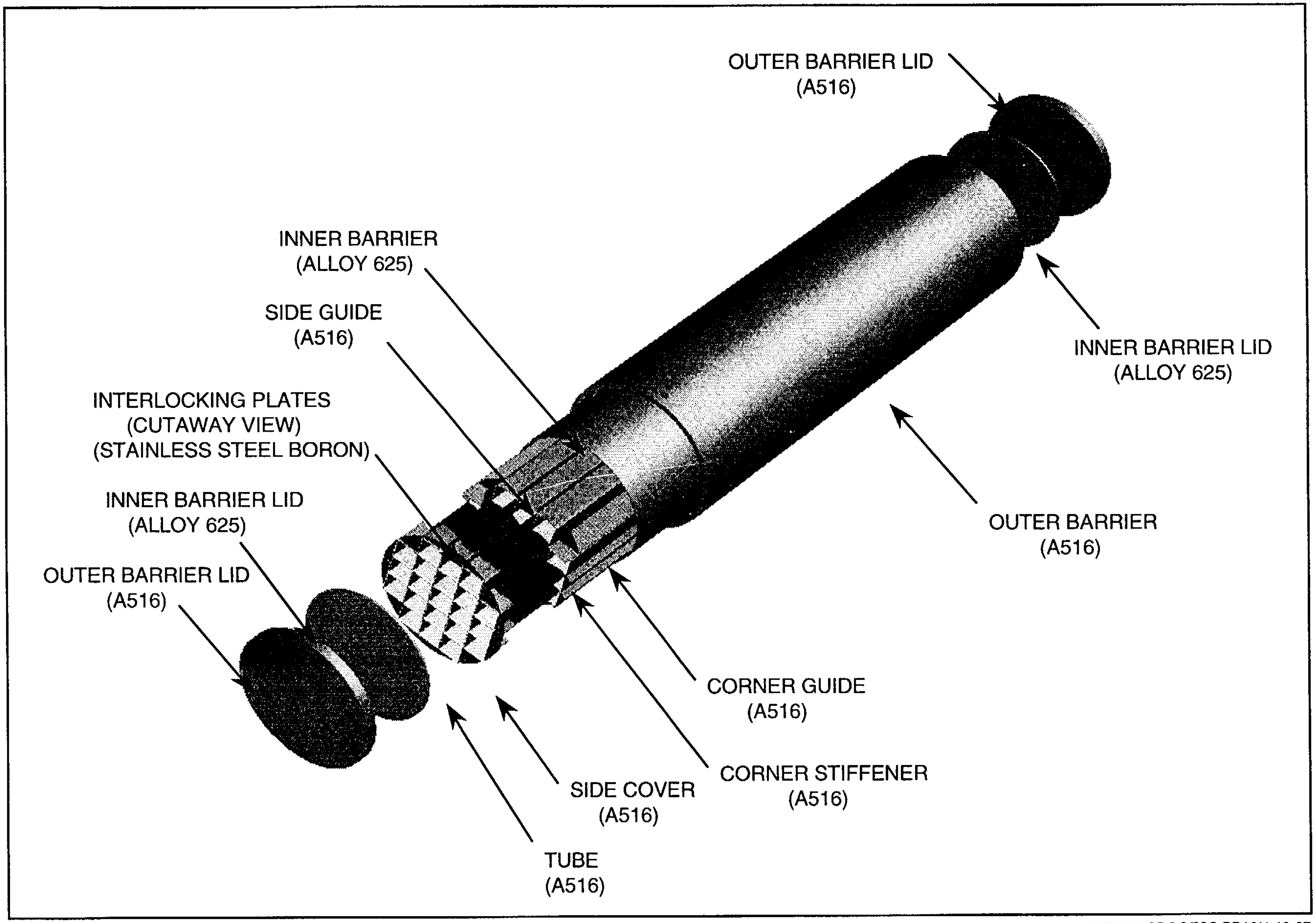
Two designs for defense waste glass are under consideration. The first, shown in Figure 5-3, would contain four waste glass canisters. The second design shown in Figure 5-4, would hold both waste glass and DOE-owned spent fuel. This second design provides for five high-level waste glass canisters surrounding 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.

<u>Materials Selection</u>

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.



PR16_5-1.PPT.125.PROGESS.PR16/4-10-97

Figure 5-1. Schematic of Disposal Container for Uncanistered Pressurized Water Reactor Spent Nuclear Fuel

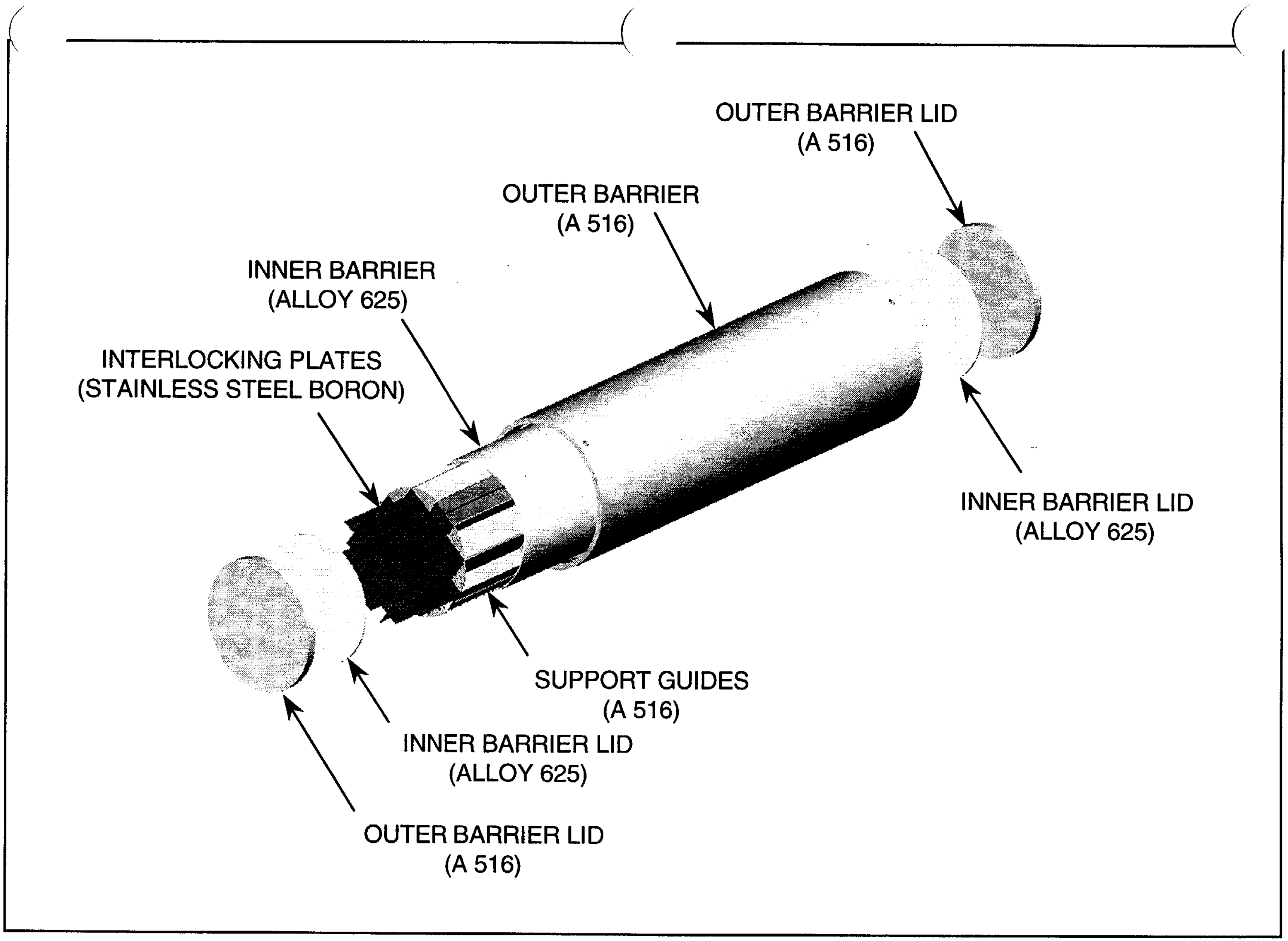


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Figure 5-2. Schematic of Disposal Container for Uncanistered Boiling Water Reactor Spent Nuclear Fuel

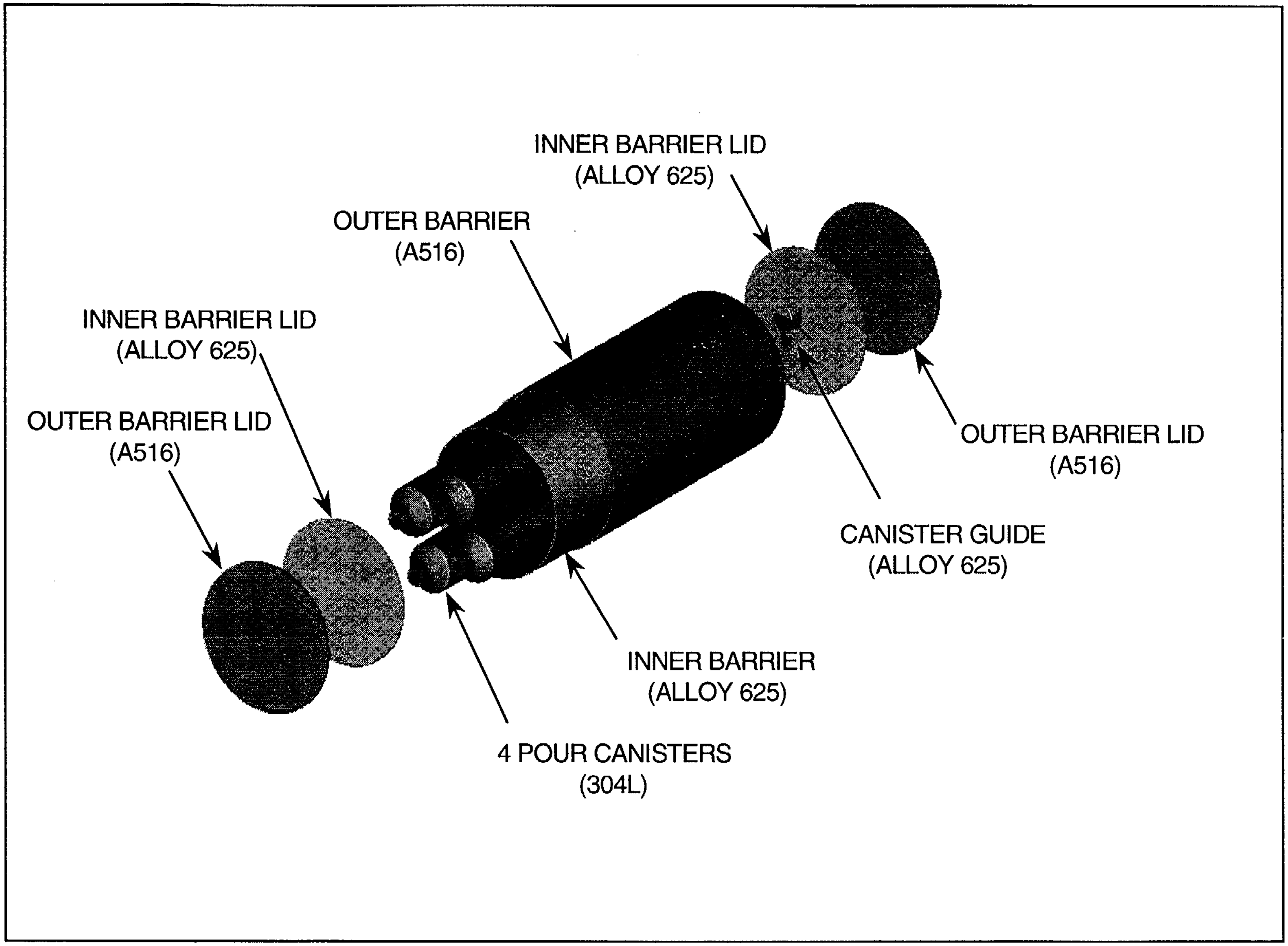

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Figure 5-3. Schematic of Disposal Container for Defense High-Level Waste Glass

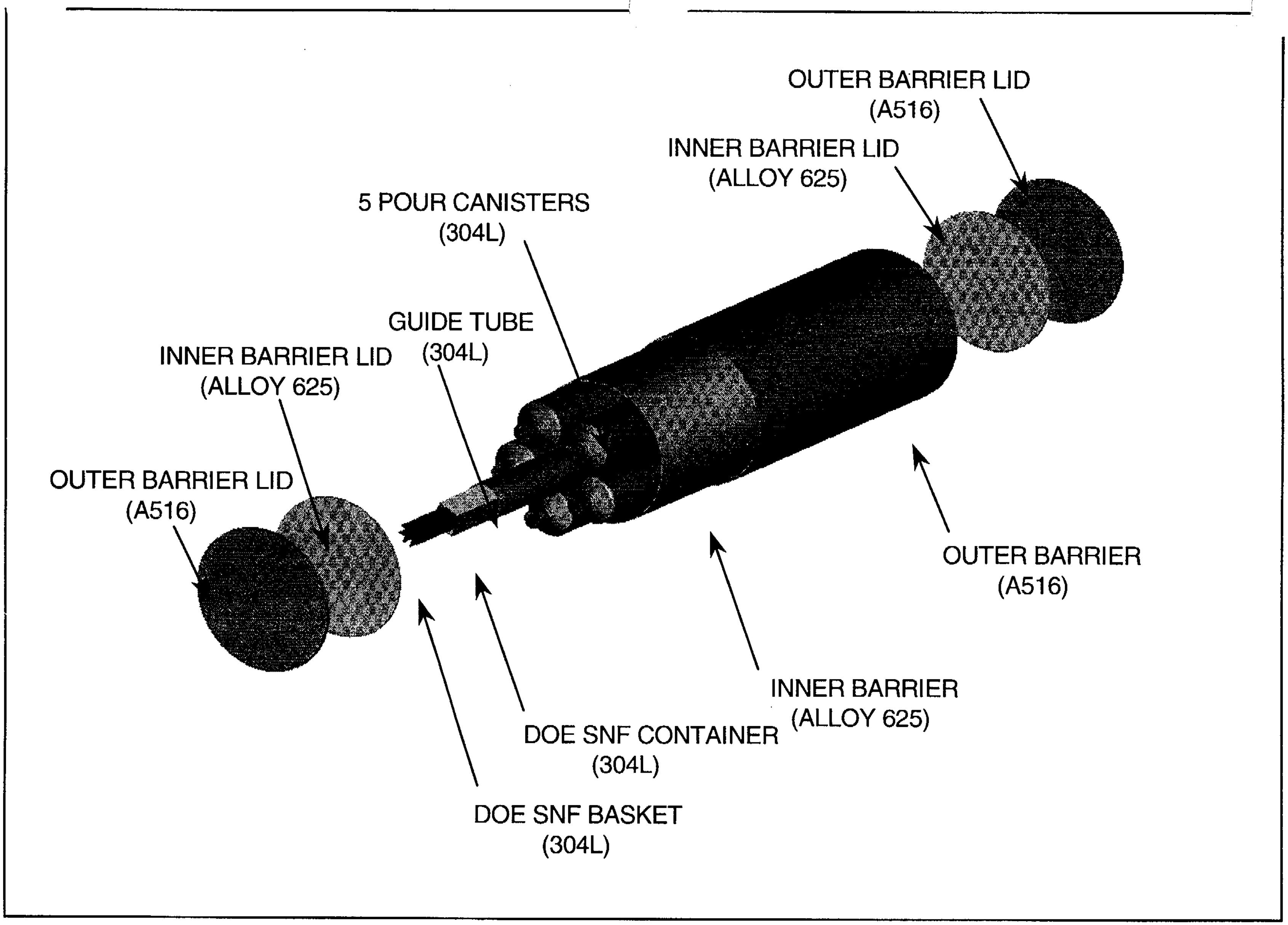

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Figure 5-4. Schematic of Disposal Container for Defesne High-Level Waste Glass and DOE-Owned Spent Nuclear Fuel

- 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 in this reporting period. One activity focused on the preliminary design of the waste package support 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 SECTION 8.3.4.2)

5.1.1 Design Activity 1.10.2.1 - Waste Package 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 exists. Even if the repository thermal load is kept below 85 MTU/acre as discussed in Section 4.1.6 of this progress report, 19 Kw waste package heat load could still represent a performance problem.

Criticality Options for Design Basis Fuel

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

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[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

	Design Basis Heat Range (W)		Design Basis Criticality Range		Coverage Range	
Waste Package Types:	Hmin	Hmax	k_∞min	k_∞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.

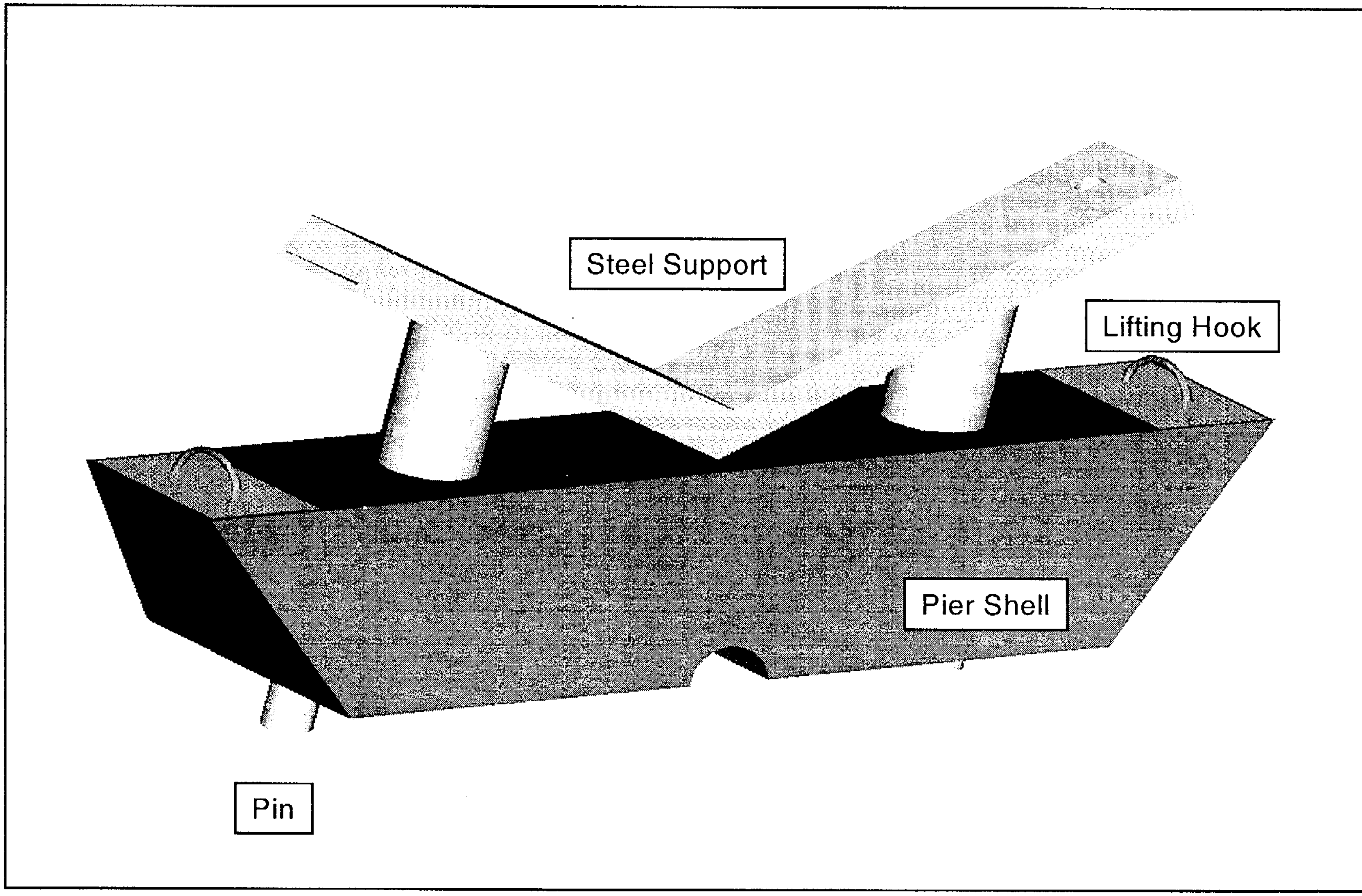

Figure 5-5. Waste Package Support Concept

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 will 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 significantly above the assumed design goals (limits). An important caveat for the previous evaluations 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.

Waste Package Structural Analyses

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. MRKJ. 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).

<u>Source Terms Analysis for Design Basis Events</u>

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 total dose per unit waste form for at least one organ. Those nuclides specified in NRC 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.

Forecast: 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 Design Activity 1.10.1.1 - Consideration of 10 CFR 60.135(a) Factors

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 hybriding, 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.

New Drift Seepage Model

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 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 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 this activity 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 available from in situ rock masses and data from smaller samples commonly tested in the laboratory do not provide information on fracture behavior. Moreover, in tests at this scale known boundary and environmental conditions can be imposed 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 38 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.

Support of the Large Block Test

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.

Fracture 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 phase of the large block test will continue throughout most of the next reporting period. Once the interior block temperature has reached approximately 140°C, with the temperature at the top 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 6.9.1 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.

<u>Activity 1.10.4.5.8 - Colloid stability experimental studies at ambient and elevated temperatures</u>. 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.

<u>Activity 1.10.4.5.9 - Biodegradation stability experimental studies at ambient and elevated temperatures</u>. 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 (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 including 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 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.

Forecast: 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 6.16 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 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. 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, 1997v) were re-interpreted 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 Report #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 "abstractions" 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 the 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 discussion of the activities currently ongoing to produce abstracted models. Also presented are the specific 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?

PROGRESS REPORT #16

Top Priority Issues for 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
    - Cathodic 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).

PROGRESS REPORT #16

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.

PROGRESS REPORT #16

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?

Synopsis of Analysis Plans for Addressing Top-Ranked Thermohydrology Issues

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 for 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 envi- ment 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 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. 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).

<u>Subactivity 1.4.2.4.1 - Assessment of degradation modes affecting ceramic-metal systems</u>. Please refer to Section 6.9.2 of this progress report for discussion on the evaluation of ceramic coatings on steel substrates.

<u>Subactivity 1.4.2.4.2 - Laboratory test plan for ceramic-metal systems of the alternate barriers investigations</u>. Please refer to Section 6.9.2 of this progress report for discussion on the evaluation of ceramic coatings on steel substrates.

<u>Subactivity 1.4.2.4.3 - Assessment of degradation modes affecting bimetallic/single metal systems</u>. 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 issue than they are for corrosion resistant materials.

Subactivity 1.4.2.4.4 - Laboratory test plan for bimetallic/single metal material system.

Long-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^{TM}-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 test 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 $\text{ksi}\cdot\text{in}^{1/2}$. 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.

Cyclic potentiodynamic polarization experiments at 60°C and 90°C using several corrosion-allowance and corrosion-resistant alloys in aqueous solutions containing 6 wt% ferric chloride ($FeCl_3$) have begun. 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.

<u>Forecast:</u> 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).

<u>Galvanic Corrosion Studies</u>

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 needs 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.

Subactivity 1.5.2.1.2 - Oxidation of spent nuclear fuel. 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 performance 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 bound the radionuclide source term.

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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 fractured cores will be placed in a core-flow device.

Forecast: The experimental measurement 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 geohydrology, 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 <u>Tracers, Fluids, and Materials (excluding water)</u>

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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CHAPTER 7 - EXPLORATORY STUDIES FACILITY DESIGN AND CONSTRUCTION

INTRODUCTION

The Exploratory Studies Facility (ESF) is being designed and constructed to allow the performance of a program of in situ exploration and testing above, at, and below the depths at which waste might be emplaced. This work will be used to determine the suitability of the Yucca Mountain site for the construction of a potential underground high-level nuclear waste repository. This work also addresses the 10 CFR 60.15(b) requirement for in situ exploration and testing at the depths at which waste would be emplaced. Generally, the ESF will provide access to underground tuff horizons to obtain technical data.

For discussion purposes, ESF work can be divided into two parts. The first part is design of the equipment and processes necessary to perform the ESF functions. Basically, this is design to support excavation activities, including design of utilities and support equipment. The second part is the construction and excavation that implements the design. This chapter reports progress in these areas. Testing and analysis activities conducted in the ESF during this reporting period are described in Chapters 3 and 5 of this progress report.

Background

> Title I refers to preliminary design, Title II refers to the definitive design that follows Title I, and Title III refers to construction engineering design that provides sufficient detail to support field work.

The Site Characterization Plan, (DOE, 1988), described an Exploratory Shaft Facility that would be used to gather site data through testing in a localized subsurface facility accessed by two vertical shafts. This facility would have been located in the northeast quadrant of the primary area identified for potential repository construction. In consultation with the U.S. Nuclear Regulatory Commission and the Nuclear Waste Technical Review Board, U.S. Department of Energy (DOE) management subsequently decided to study alternative configurations for the ESF and the Geologic Repository Operations Area. The results were documented in a report entitled ESF Alternatives Study: Final Report (Dennis, 1991). One alternative configuration, called Option 30, was judged to be the most desirable overall concept. After several minor modifications, detailed in a document entitled Documentation of the Evaluation of Findings of the ESFAS Used to Develop a Reference Design Concept (DOE, 1991), the revised Option 30 configuration was used to form the basis for a revised Title I ESF design effort. At that time, the facility was renamed the Exploratory Studies Facility. The Title I design was described in the Title I Design Summary Report for the Exploratory Studies Facility (DOE, 1992a). This configuration was also described in Sections 1.3.1 and 2.1.2 of Progress Report #8 (DOE, 1993b).

Title II ESF design began in fiscal year (FY) 1992. Early activities centered on the design of north portal surface facilities. Underground ESF Title II design began in FY 1993, concurrent with the start of repository advanced conceptual design. Early in repository and Title II ESF design, the ESF Option 30 layout was adjusted. This adjustment resulted in access ramps with flatter grades to support underground facility access by rail, flat-lying emplacement drifts, and emplacement drift arrangements that to a great extent avoid the Ghost Dance fault. The change to this configuration was detailed in Section 2.1.9.3 of Progress Report #9 (DOE, 1994g).

Title II ESF design was divided into design packages to allow phased design and construction. These design packages are referenced throughout Chapter 7. Where applicable, design elements have been subdivided further to meet the needs of the constructor. Plans for development of design packages for the Thermal Test Area adjacent to the north ramp extension, Calico Hills level north ramp, and Calico Hills level main and cross drifts were deleted with the implementation of the Civilian Radioactive Waste Management Program Plan, Revision 1 (DOE, 1996a). Table 7-1 gives the package identification, lists the elements of each design package, and provides the completion status of each package as of the end of this reporting period. Figure 7-1 shows the general physical locations within the ESF of the work supported by the design packages.

A common nomenclature was previously established for existing and planned ESF alcoves. This nomenclature will be used for testing and other purposes. Previously, alcoves were identified by planned physical construction sequence. The new identification system replaces the numerical identifier system with a functional test-based nomenclature. The new ESF alcove names, given in Table 7-2, have been incorporated into the design package identification and descriptions provided in Table 7-1 and will be used in future documentation.

Summary of Activities During this Reporting Period

The main focus of ESF work during this reporting period was to continue tunnel boring machine excavation and test alcove excavation, while constructing only the surface support facilities necessary to support subsurface construction. By the end of this reporting period, tunnel boring machine excavation was 132 m and 32 calendar days behind the schedule promulgated in Revision 1 of the Civilian Radioactive Waste Management Program Plan (DOE, 1996a). The excavation was behind schedule because of less-than-favorable ground conditions encountered in the south ramp.

Design work continued on the Thermal Testing Facility and the Northern Ghost Dance Fault Alcove. Designs for the excavation of the Thermal Testing Facility heated drift and associated test support features and the drill/test room for the Northern Ghost Dance Fault Alcove were issued for construction.

Table 7-1. Design Package Identification and Description

Design Package	Design Package Description	Status
1	**North Portal Site Preparation and Surface Facilities**	
1A	Elements include • North portal pad • Topsoil storage area • ESF access road • Sewage collection and treatment system • North portal pad water supply system • Tunnel boring machine starter tunnel • Rock storage area • Switchgear building • Modifications to switchgear building for the integrated data and control system • North portal pad power distribution system (partial) • Local controls for water system	Issued
1B	Elements include • Change house building • Shop building • Sanitary sewer system • Power distribution system • Water distribution system • Subsurface wastewater system • H-road, site grading, and paving • Site grounding • Redesigned north portal pad drainage • Reconfiguration of north portal pad for surface rail • Completion of grading and paving plan and surface rail • North portal pad perimeter fencing • North portal lightning protection	Issued
1C	Elements include: • Compressed air system • Standby power system • Site lighting (partial) • Site grounding (partial)	Issued

Table 7-1. Design Package Identification and Description (continued)

Design Package	Design Package Description	Status
1D	Elements include • Site grounding (continuation) • Muck storage area • Conveyor maintenance access road • Fuel storage system • Site lighting (continuation) • Equipment foundations • Compressed air system condensate collection system	Issued
1E	Elements include • Standby/auxiliary power • Site grounding (continuation) • ESF electrical distribution (continuation) • Revised fuel storage system	Issued
Integrated Data and Control System	Elements include • Procurement specification for the complete integrated data and control system • Fiber-distributed data interface for first phase of integrated data and control system • Data Collection System	 Issued Issued FY 1997
2	**North Ramp Excavation - Starter Tunnel to Topopah Spring Level**	
2A	Elements include • Surface and subsurface conveyor • Electrical switchgear, transformers, and power centers	Issued
2B	Elements include • Utility systems • Tunnel ventilation specifications and drawings • Rail haulage system • Mapping platform procurement specification • Excavation, ventilation, and muck storage trade studies • Control system specification and drawings • Life safety monitoring and warning system	Issued

Table 7-1. Design Package Identification and Description (continued)

Design Package	Design Package Description	Status
2C	Elements include • North ramp to Topopah Spring level excavation and ground support • Remaining utility systems for north ramp • Electrical power, lighting, and grounding • Associated test and operational support alcoves	Issued
(a)	Topopah Spring level main drift	Issued
(a)	Ghost Dance fault accesses (at Topopah Spring level) • Northern Ghost Dance Fault Alcove (Alcove #6) - Northern Ghost Dance Fault Access Drift (first 95 meters of excavation) - Northern Ghost Dance Fault Drill/Test Room (remaining 110 meters of excavation) • Southern Ghost Dance Fault Alcove (Alcove #7) - Southern Ghost Dance Fault Access Drift (first 120 meters of excavation) - Southern Ghost Dance Fault Drill/Test Room (remaining approximately 60 meters of excavation)	 Issued Issued Issued FY 1997
(a)	North ramp extension (Referred to as "ESF East-West Drift" in Draft Civilian Radioactive Waste Management System Program Plan, Revision 1 and planned for FY 1999. The DOE recently accelerated schedule to FY 1999)	FY 1998
(a)	Thermal testing area • Thermal Testing Facility (Alcove #5) - Thermal Testing Facility (access/observation drift) - Thermal Testing Facility (thermomechanical alcove) - Thermal Testing Facility (cross drift to beginning of heated drift) - Thermal Testing Facility (heated drift) * Layout and Ground Support * Thermal Bulkhead * Invert * Cast-in-place Liner Test System * Drill Bay - Cable Trays - Power Distribution System - Ventilation System	 Issued Issued Issued Issued FY 1997 Issued Issued Issued Issued FY 1997 FY 1997

Table 7-1. Design Package Identification and Description (continued)

Design Package	Design Package Description	Status
(a)	Topopah Spring level south ramp	Issued
(a)	South portal pad and facilities - South portal pad - South portal highwall and boxcut - South portal lightning protection system - South portal ventilation system	 Issued Issued Issued FY 1997
New	Drift scale flux test niches	FY 1997

(a) The package number designations for these packages were previously deleted as discussed in Progress Report #12 (DOE, 1995g). Since then, these packages have been referred to by their names only.

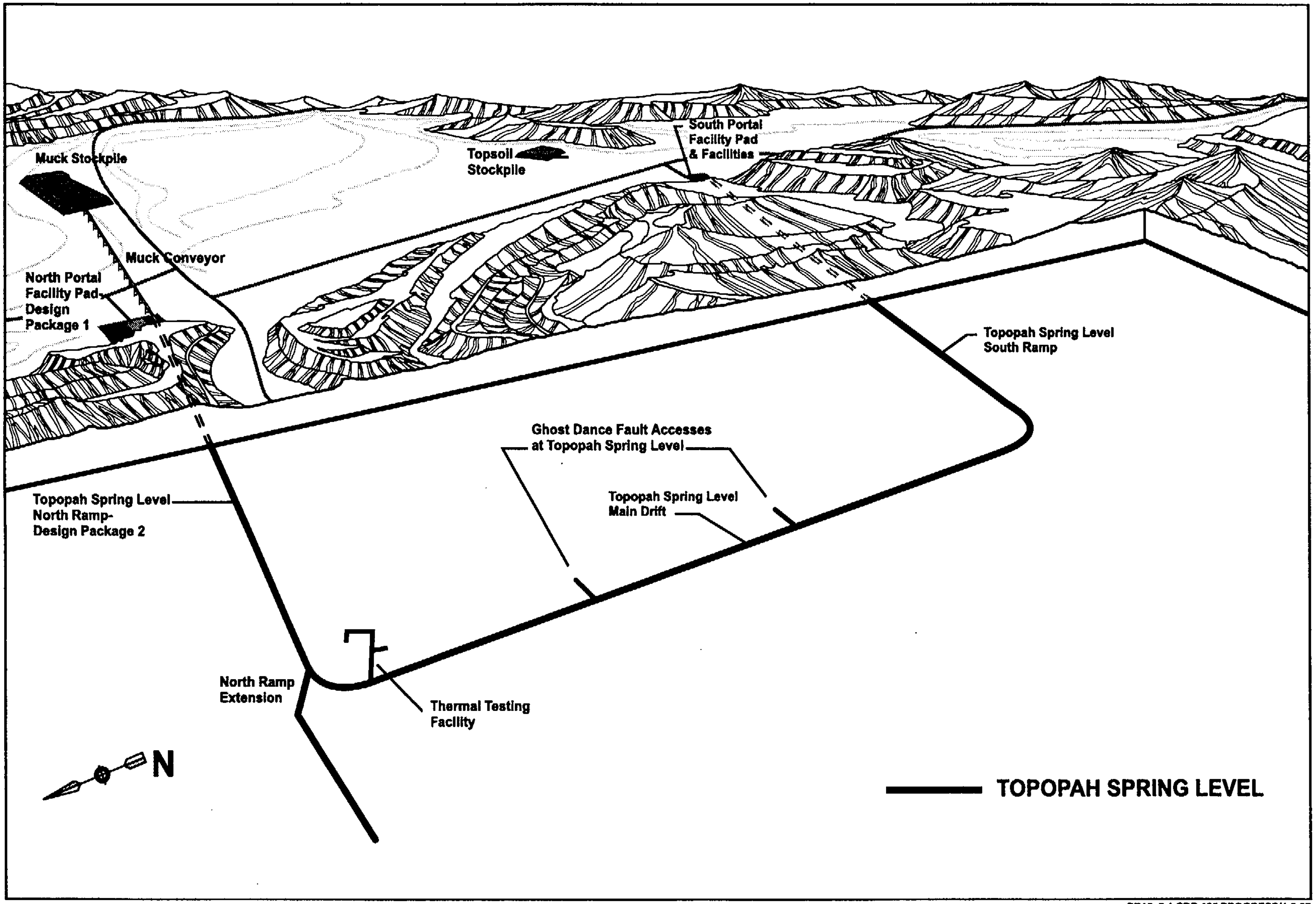

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Figure 7-1. Exploratory Studies Facility Design Package Locations

Table 7-2. Exploratory Studies Facility Nomenclature

Alcove name	Accepted alcove acronym	Construction sequence alcove number (for informal reference)
Upper Tiva Canyon Alcove (Anisotropic)	UTCA	Alcove #1
Bow Ridge Fault Alcove	BRFA	Alcove #2
Upper Paintbrush (Non-Welded) Contact Alcove	UPCA	Alcove #3
Lower Paintbrush (Non-Welded) Contact Alcove	LPCA	Alcove #4
Drill Hole Wash Fault Alcove (Deferred)	DWFA	Deferred
Thermal Testing Facility	TTF	Alcove #5
Northern Ghost Dance Fault Alcove	NGDFA	Alcove #6
Southern Ghost Dance Fault Alcove	SGDFA	Alcove #7

The Yucca Mountain Site Characterization Project (Project) initiated a systematic process in the second quarter of FY 1997 to determine the activities necessary to complete construction of the 25 ESF systems. These evaluations may conclude that (a) the system must be completed to the approved design, (b) the system as currently constructed is acceptable, or (c) the system requires additional upgrading to a configuration between the currently constructed design and the approved design. If necessary, the ESF Design Requirements (DOE, 1996h) and the as-built design documentation will be updated to capture the final as-built field configuration and associated requirements.

Construction of underground support and utility facilities continued at a rate needed to support the progress of the tunnel boring machine. Excavation of the Thermal Testing Facility was completed in February 1997, which satisfied a Project milestone. Also, Phase I excavation (Northern Ghost Dance Fault access drift) of the Northern Ghost Dance Fault Alcove was completed. The initial phase (approximately 134 m of excavation) of the Southern Ghost Dance Fault access drift portion of the Southern Ghost Dance Fault Alcove was completed.

7.1 EXPLORATORY STUDIES FACILITY DESIGN

ESF design activities proceeded in support of ESF construction and tunnel boring machine operations. The following subsections describe the initiatives and other activities on the various design packages that occurred during this reporting period.

7.1.1 Quality Assurance Activities

Two specifications covering the maintenance and operations of surface facilities and the storage, handling, and operation of materials and equipment were revised to capture the changes made to requirements in the latest revision of the Determination of Importance Evaluation for Surface ESF. The specification revisions maintain traceability to determination of importance evaluation requirements, clarify inspection requirements, and ensure that field operations will have minimum impact on the site and ongoing testing activities. Table 7-3 summarizes ESF design quality assurance activities during this reporting period.

Table 7-3. Exploratory Studies Facility Deficiencies

Graded Deficiencies Received by the Exploratory Studies Facility Design Organization - Issuing Organization	Deficiencies Outstanding as of 10-1-96	Deficiencies Outstanding as of 3-31-97
Corrective Action Requests (most significant)	None	None
Deficiency Reports - Yucca Mountain Quality Assurance Division	4	3
Deficiency Reports - Management and Operating Contractor Quality Assurance	1	1
Performance Reports (least significant--isolated) - Yucca Mountain Quality Assurance Division	0	0
Performance Reports (least significant--isolated) - Management and Operating Contractor Quality Assurance	1	0
TOTALS	**6**	**4**

7.1.2 Design Progress

The following paragraphs report progress made during this reporting period in completing ESF design.

Design Package 1 (North Portal Site Preparation and Surface Facilities)

Two design modifications for Design Package 1 were completed, baselined, and issued for construction by Engineering Change Request: (1) clarification of the specification requirements covering asphalt/concrete surface courses and (2) clarification of the specification requirements covering the sanitary sewer collection system.

Integrated Data and Control System

The data collection system portion of the integrated data and control system equipment purchase order was delivered to the Project in March 1997. The system consists of approximately 10,000 channels of data acquisition hardware housed in 5 cabinets, capable of collecting, storing and transferring data collected from instruments that will be installed in the drift-scale test. The cabinets are currently being configured to accept the types of instrument outputs that will feed data to them.

Design Package 2 (North Ramp Excavation - Starter Tunnel to Topopah Spring Level)

The ESF North Portal Stability Analysis (CRWMS M&O, 1997w) was completed and issued. This analysis examined the stability of existing structural systems at the ESF north portal boxcut and starter tunnel opening. The analysis presents two alternatives for future work in this area (a permanent replacement structure at the north portal or additional testing and analysis of the boxcut and starter tunnel) to ensure the integrity of the system under in situ loadings and possible seismic events.

Topopah Spring Level Main Drift and South Ramp

The analysis for ground support for alcoves was revised. The revision of this analysis established the extent of shotcrete required at the intersections of the ESF main loop and the various test alcoves to supplement the existing rockbolt and steel set ground-support systems to ensure long-term stability of the facility. This revision also removed a large number of To Be Verified (TBV) issues found in the previous revision. Application of shotcrete in the tunnel will begin after tunnel boring machine exits the south portal to avoid interference with ongoing construction activities.

South Portal Pad and Facilities

Final designs for the south portal and pad were released in the previous reporting period. ESF design personnel continued to monitor ongoing construction of the south portal to ensure that the actual ground conditions encountered were within the assumptions made for the design analysis. ESF design personnel were also responsible for monitoring the blasting activities at the south portal.

The design package for the south portal lightning system was completed and issued for construction. This system will provide for personnel protection and will be in place before the tunnel boring machine is removed from the tunnel.

Thermal Testing Facility [at Station 28 + 27 (2827 m)]

The Thermal Testing Facility is being designed in phases to meet the needs of the testing organizations. The following design activities for the heated drift portion of the Thermal Testing Facility were completed in this reporting period.

- Layout and Ground Support Drawing Package. This design package covered the general arrangement of the drift excavation and special ground support details.

- Cast-in-place Liner Test System Design Package. The cast-in-place liner will become part of the drift scale test and will allow for monitoring of the effects of high heat loads on concrete liners in an environment similar to the potential repository emplacement drifts during the test. The data collected will be crucial in any future designs for structural liners required in the potential repository. The cast-in-place liner design package consisted of a design analysis and two detail drawings of the liner. The design analysis was a cooperative effort with the Repository Design Group. The analysis provides the basis for selection of critical cast-in-place lining construction, configuration, materials, and controls to ensure that the test results can be qualified for use as a design input of potential repository lining systems.

- Concrete Invert design drawings. This design package covered detailed designs for the concrete floor system required in the heated drift.

- Cable Tray Installation Sketches. A set of four sketches showing the cable tray installation inside and outside the heated drift was transmitted to the ESF Test Coordination Office for inclusion in the appropriate Field Work Package. While not a formal design package, these sketches were prepared to assist the testing organization in developing the arrangement of the Thermal Testing Facility.

Northern Ghost Dance Fault Alcove [at Station 37 + 37 (3737 m)]

The design drawings for the Northern Ghost Dance Fault Alcove were revised to incorporate the Northern Ghost Dance Fault Drill/Test Room and issued for construction near the end of this reporting period. This revision included plan and section drawings for the extension of the drift though the fault and creation of a drill/test room on the far side of the fault. Completion of this excavation will support ongoing testing of the Ghost Dance fault structure.

Southern Ghost Dance Fault Alcove [at Station 50 + 64 (5064 m)]

Design of a drift extension and drill/test room, similar to that designed for the Northern Ghost Dance Fault Alcove, will be completed later in FY 1997.

Other Design Activities

A Technical Document Preparation Plan for the Construction Completion Evaluations on the alternate construction utilities was completed and issued. Alternate construction utilities are those utility systems (e.g., power, compressed air, lighting, ventilation, etc.) that were designed and installed by the constructor to support excavation of the ESF. These evaluations will examine each alternate construction utility system of the ESF and will define the minimum design and/or construction modifications required for these systems to continue supporting the requirements of ongoing ESF testing.

Forecast: The following activities are scheduled for second half of FY 1997:

- Approval and issuance for construction of the south portal "flow-through" ventilation system. This system will be designed as a modification to the existing construction ventilation system to support ongoing testing in the ESF after tunnel boring machine exit at the south portal.

- Approval and issuance for construction of drawings and specifications for the remaining portion of the Thermal Testing Facility. Areas of work remaining include completion of the thermal bulkhead, ventilation system, and power distribution system designs.

- Revision of the design drawings for the Southern Ghost Dance Fault Alcove to include the drift extension and planned drill/test room layouts.

- Completion of configuration of the data collection system cabinets and shipment to the ESF north portal pad for installation, scheduled for May 1997. Wiring of the instruments to the data collection system and pre-operational checkout are scheduled to begin in June 1997.

- Completion of 13 Architect/Engineering Construction Completion Evaluations with associated as-constructed documentation and system descriptions. These evaluations will examine each alternative construction utility system of the ESF and will define the minimum design and/or construction modifications required to ensure these systems to continue supporting the ongoing ESF testing program. The construction report for the "Q" ground support will begin in FY 1997 but will not be completed until FY 1998.

- Continuation of the ground support confirmation process. The ESF ground-support systems will be evaluated to assess and confirm their adequacy. These evaluations will be based on the comparison of the design with the results of ESF construction monitoring. Confirmation reports covering ground support up to Station 56 + 00 (5600 m) are scheduled to be released in FY 1997.

- Support of completion of ESF change house construction. Any additional design document revisions required to allow completion of the ESF change house will be prepared.

- Approval and issuance for construction of design layout drawings for the drift-scale flux test niches. The niches will be designed to support testing requirements. The niches, due to location on the right rib of the Topopah Spring main drift, will be located to minimize impact on potential repository emplacement drifts.

7.2 EXPLORATORY STUDIES FACILITY SEISMIC DESIGN

The purpose of this activity is to report on progress in ESF seismic design.

No further work occurred during this period in this area. Seismic design basis requirements for the ESF permanent and temporary items are documented in Appendix A of the ESF Design Requirements (DOE, 1996h). Seismic design basis requirements applicable to ESF permanent items are shown in this document as "to be verified." ESF items were designed using the appropriate seismic design basis requirements in the ESF Design Requirements.

Forecast: No activities are planned in FY 1997. Over the long term, engineering analysis based on the methodologies of the two seismic topical reports developed to date will be used to verify the seismic design basis requirements for the permanent items provided in the ESF Design Requirements document. The engineering analysis and the removal of the "to be verified" items from the ESF Design Requirements document are expected to be completed in FY 1998. Upon completion of this task, the design of the ESF permanent items will be evaluated for any impacts.

7.3 EXPLORATORY STUDIES FACILITY CONSTRUCTION

Significant progress was made on the construction of the ESF. Figure 7-2 provides a schematic view of tunnel boring machine progress as of March 31, 1997. The "scheduled" progress in Figure 7-2 reflects the schedule promulgated in Revision 1 of the Civilian Radioactive Waste Management Program Plan (DOE, 1996a). This schedule was revised at the time the Program Plan revision was made to reflect the very large distance by which the tunnel boring machine was ahead of the previous schedule.

Less-than-favorable ground conditions during December and January of this reporting period resulted in slower than expected ESF tunnel advancement. The Northern Ghost Dance Fault Alcove was under construction at the end of the reporting period, along with the Topopah Spring level south ramp.

7.3.1 Tunnel Boring Machine Operations

The tunnel boring machine continued excavating in the TSw1, crystal-poor upper lithophysal unit of the Topopah Springs unit, until encountering the Tiva Canyon, Tpcpv, at approximately Station 74 + 37 (7437 m) in late February 1997. The tunnel boring machine is expected to remain in the Tiva Canyon unit until it exits at the south portal at Station 78 + 78 (7878 m).

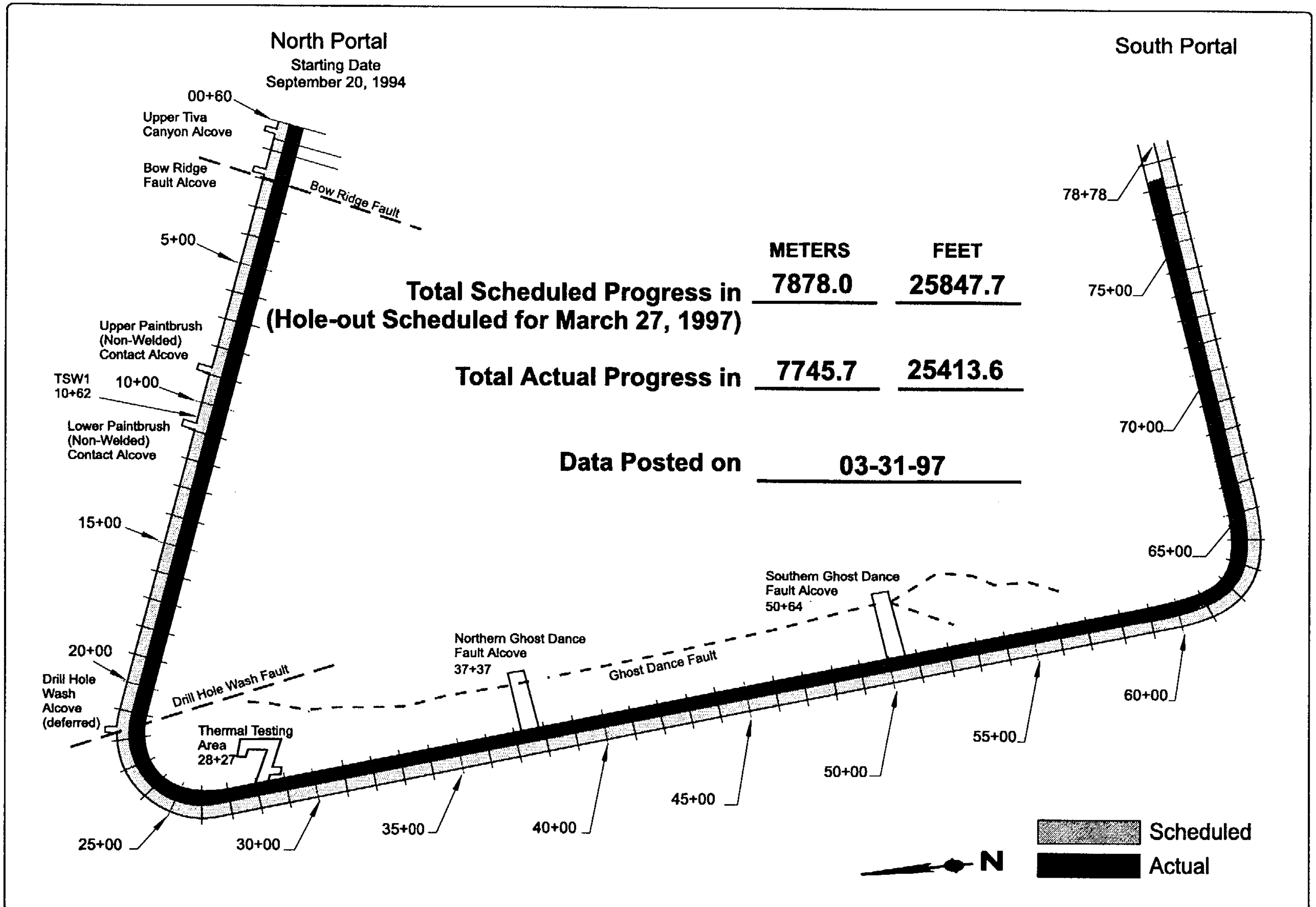

Figure 7-2. Tunnel Boring Machine Progress

At the beginning of this reporting period, the tunnel boring machine was at Station 64 + 78 (6478 m), which was 174 m and 27 calendar days ahead of the revised Program Plan schedule. The tunnel boring machine advanced 1268 m during this reporting period and at the end of the period was at Station 77 + 46 (7746 m), 32 calendar days and 132 m behind schedule. During this reporting period, 42 m and 59 calendar days were lost against the revised Program Plan schedule. As mentioned in Progress Report #15, the project adopted a new Project baseline in June
1996. This 1996 baseline schedule placed the tunnel boring machine exit date at March 27, 1997.

During the first two months of this reporting period, the tunnel boring machine advanced at an average rate of 21.6 m per excavation day until highly fractured rock was encountered in late November that significantly reduced progress. During December and January the average advance rate fell to 2.5 m per excavation day in highly fractured rock that required extensive hand mucking of material around the tunnel boring machine before the installation of invert segments and also required almost continuous steel sets for ground support. In early February ground conditions improved significantly, allowing the tunnel boring machine advance rate to average 20 m per excavation day for the month.

7.3.2 Tunnel Boring Machine 500-Hour Maintenance

At the beginning of the first quarter of FY 1997, the tunnel boring machine was shut down for seven days to perform the fourth 500-hour maintenance. The tunnel boring machine is equipped with an hour meter that operates while the cutter head is rotating; this meter is used to determine the 500 hours of machine use. The maintenance was performed to ensure continued reliable operation and to meet the manufacturer's warranty requirements. As with the previous 500-hour maintenance activities, the fourth 500-hour maintenance generally revealed conditions expected for equipment subjected to the type of material that has been encountered. The main requirement was for hard facing and installing wear plates on the cutter head.

7.3.3 Test Alcove Construction

Test alcoves are being excavated to provide the project with dedicated testing areas free of construction influences to acquire data on the geologic, hydrologic, geochemical, and geothermal characteristics of the site. These alcoves are located at predetermined sites along the north ramp and the Topopah Spring level main drift. The general locations correspond to specific test requirements (such as the need to investigate a fault structure). The specific locations chosen are those expected to provide maximum relevant test data acquisition while minimizing test interference. The tests conducted in these alcoves are described in Chapters 3 and 5 of this progress report.

Three test alcoves were under construction during the reporting period: the Thermal Testing Facility located at Station 28 + 27 (2827 m), the Northern Ghost Dance Fault Alcove located at Station 37 + 37 (3737 m) and, the Southern Ghost Dance Fault Alcove located at Station 50 + 64 (5064 m). Excavation of the Thermal Testing Facility was completed in early February 1997. Completion of excavation of the Thermal Testing Facility satisfied the Program milestone to complete excavation of the alcove. Construction is continuing in the Thermal Testing Facility to support the planned drift-scale test scheduled to start in the first quarter of FY 1998.

Characterization of the Northern Ghost Dance Fault Alcove was completed during this reporting period, and Phase II (Northern Ghost Dance Fault Drill/Test Room) excavation resumed in March 1997.

Construction of the Southern Ghost Dance Fault Alcove began in October 1996 and was completed to Station 1 + 34 (134 m) in late February 1997. At this time, excavation was halted to allow for planned test drilling and instrumentation to be installed to support fault characterization. Excavation to complete Phase I (Southern Ghost Dance Fault Access Drift) of the Southern Ghost Dance Fault Alcove resumed in mid-March 1997.

7.3.4 Surface Facilities and Utilities Construction

Construction of the change house on the north portal pad, which was deferred during the first quarter of FY 1996 because of budgetary constraints, resumed during this reporting period.

The south portal road, pad, and boxcut construction was completed during the reporting period. This task included stripping the topsoil for the road and the pad, excavating the boxcut, constructing the demobilization pad for the tunnel boring machine, and setting up the utility needs for the demobilization of the tunnel boring machine operation.

Forecast: The rate of tunnel boring machine advancement is expected to decrease as it nears the ground surface and encounters increasingly weathered rock resulting in an increased need for installation of steel set ground support. The tunnel boring machine is scheduled to exit at the south portal early in the third quarter of FY 1997.

Phase II (Northern Ghost Dance Fault Drill/Test Room) excavation of the Northern Ghost Dance Fault Alcove is scheduled to be completed in the third quarter of FY 1997.

Phase I (Southern Ghost Dance Fault Access Drift) excavation of the Southern Ghost Dance Fault Alcove is scheduled to be completed in the third quarter of FY 1997. Phase II (Southern Ghost Dance Fault Drill/Test Room) excavation of the Southern Ghost Dance Fault Alcove is scheduled to start in the fourth quarter of FY 1997 and be completed in the first quarter of FY 1998.

Completion of conversion of the tunnel ventilation system after tunnel boring machine exit and construction of the change house on the north portal pad is scheduled for the fourth quarter of FY 1997.

Completion of the switchgear building and potable water system on the north portal pad and the north portal pad drainage is scheduled for FY 1998.

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